DOE/EIS-0218F
February 1996

Summary

Final Environmental Impact Statement

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Department of Energy
Washington, DC 20585

February 8, 1996

Dear Interested Party:

I am enclosing a copy of the Summary of the final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. The Department of Energy, in cooperation with the State Department, prepared the final Environmental Impact Statement.

This study analyzes the potential environmental impacts of adopting a policy to manage foreign research reactor spent fuel containing uranium enriched in the United States. In particular, the study examines the comparative impacts of several alternative approaches to managing the spent fuel. The analyses demonstrate that the impacts on the environment, workers and the general public of implementing any of the alternative management approaches would be small and within applicable Federal and state regulatory limits.

The Department's preferred approach to managing the spent fuel, referred to in the study as the "preferred alternative," is for the Department to receive the spent fuel into the United States, and to manage it at the Department's Savannah River Site in South Carolina and the Idaho National Engineering Laboratory. The spent fuel would be shipped to the United States over 13 years through two military ports. The Charleston Naval Weapons Station in South Carolina would receive about one to two shipments every month beginning in 1996. The Concord Naval Weapons Station in California would receive far fewer shipments (as few as five shipments over a 13-year period) beginning in 1997.

The final Environmental Impact Statement is a three-volume document, approximately 4000 pages in length. Volume 1 (494 pages) describes the policy considerations of adopting a policy to manage foreign research reactor spent fuel, and the potential environmental impacts. Volume 2 (1111 pages) contains eight appendices relating to the technical analyses. Volume 3 (2230 pages) contains the public's comments on the draft Environmental Impact Statement, the Department's responses to those comments, and summaries of the 17 public hearings held throughout the United States during the 90-day comment period on the draft.

Our experience has taught us that many people who are interested in the Department's proposed activities do not necessarily want to receive a lengthy, multi-volume document to review. For this reason, we are sending you the Summary alone at this time. If, however, you would like a copy of the entire study, a particular volume, or an additional copy of the Summary, we would be pleased to send it to you. Please let us know by calling the Department's

Center for Environmental Management Information at 1-800-736-3282 (toll-free). The entire document will be placed in the public reading rooms and information locations listed in the Summary.

The Department will not make a final decision on whether to adopt the proposed policy until late March 1996. Thank you for your interest in this proposed action.

Sincerely,

Thomas P. Grumbly

Thomas P. Grumbly
Assistant Secretary for
Environmental Management

Enclosure

DOE/EIS-0218F
February 1996

Summary

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Title pg

Cover Sheet

Responsible Agencies: Lead Agency: United States Department of Energy
Cooperating Agency: United States Department of State

Title: Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Contact: For further information, concerning this Final Environmental Impact Statement, contact:

Charles Head, Program Manager
Office of Spent Nuclear Fuel Management (EM-67)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

For general information on the United States Department of Energy's National Environmental Policy Act process, call 1-800-472-2756 to leave a message, or contact:

Carol Borgstrom, Director
Office of NEPA Policy and Assistance (EH-42)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
202-586-4600

Abstract: The United States Department of Energy and United States Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, by seeking to reduce and eventually eliminate highly-enriched (weapons-grade) uranium from civilian commerce worldwide. Environmental effects and policy considerations of three Management Alternative approaches for implementation of the proposed policy are assessed. The three Management Alternatives analyzed are: (1) acceptance and management of the spent nuclear fuel by the Department of Energy in the United States, (2) facilitate the management of the spent nuclear fuel at one or more foreign facilities (under conditions that satisfy United States nuclear weapons nonproliferation policy objectives), and (3) a combination of elements from one or both of Management Alternatives 1 and 2 (Hybrid Alternative). A No Action Alternative is also analyzed.

For each Management Alternative, there are a number of implementation alternatives. For Management Alternative 1, this document addresses the environmental effects of various implementation alternatives, such as varied policy durations, management of various quantities of spent nuclear fuel, chemical separation, developmental treatment and/or packaging technologies, and differing financing arrangements. Environmental impacts are also examined at various potential ports of entry, along truck and rail transportation routes, at candidate management sites, and for alternate storage technologies. For Management Alternative 2, this document addresses the environmental effects of two implementation alternatives: (1) assisting foreign nations with storage; and (2) assisting foreign nations with reprocessing

of the spent nuclear fuel. With respect to Management Alternative 3, an example Hybrid Alternative is analyzed wherein a portion of the spent nuclear fuel would be processed at overseas facilities and the remaining portion would be managed in the United States.

The United States Department of Energy and United States Department of State, in consultation with other government agencies, designate the acceptance and management of the foreign research reactor spent nuclear fuel in the United States (i.e., Management Alternative 1 with modifications to several basic implementation elements) as the preferred alternative.

Public Comments: The public comment period on the Draft EIS was conducted from April 21, 1995 to July 20, 1995. During this period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and 5 candidate spent nuclear fuel management sites. In addition, a public hearing was held in Washington, D.C. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

Table of Contents

List of Figures

List of Tables

Summary

S.1 Introduction

Reducing the threat of the proliferation of nuclear weapons is one of the foremost goals of the United States. Proper management of spent nuclear fuel from foreign research reactors supports this goal, since much of this spent nuclear fuel contains highly-enriched uranium (HEU) which can be directly used in simple nuclear weapons.

The proposed action is for the U.S. Department of Energy (DOE) and the Department of State to jointly adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed action. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and eventually eliminate, HEU from civilian commerce.

DOE and the Department of State have evaluated various Management Alternatives for implementing this policy. A key element of DOE and Department of State decisionmaking is a thorough understanding of the policy considerations and environmental impacts that may be associated with implementation of the proposed action. The National Environmental Policy Act of 1969 (NEPA), as amended, provides Federal agency decisionmakers with a process to use in considering potential environmental impacts (both positive and negative) of proposed actions before agencies make decisions.

National Environmental Policy Act

National Environmental Policy Act of 1969: A law that requires Federal agencies to consider in their decisionmaking processes the potential environmental effects of proposed actions and analyses of alternatives and measures to avoid or minimize any adverse effects of a proposed action.

Alternatives: The range of reasonable options, including not taking any action (the No Action alternative), considered in selecting an approach to meeting the need for agency action.

Environmental Impact Statement: A detailed environmental analysis for a proposed major Federal action that could significantly affect the quality of the human environment. A tool to assist in decisionmaking, it describes the positive and negative environmental effects of the proposed undertaking and alternatives.

Record of Decision: A concise public record of DOE's decision, which discusses the decision, identifies the alternatives (specifying which ones were considered environmentally preferable), and indicates whether all practicable means to avoid or minimize environmental harm from the selected alternative were adopted (and if not, why not).

In following this process, DOE and the Department of State prepared a draft Environmental Impact Statement (EIS) for public comment. The Draft EIS was issued in April 1995. Following consideration of public comments, DOE and the Department of State have prepared this *Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (Final EIS). DOE's and the Department of State's decisions will be presented in a Record of Decision to be issued not less than 30 days after issuance of the Final EIS.

S.1.1 Policy Background

Since the 1950's, as part of the "Atoms for Peace" program, the United States has provided peaceful nuclear technology to foreign nations in exchange for their promise to forego development of nuclear weapons. A major element of this program was the provision of research reactor technology and the HEU necessary to fuel the research reactors. Research reactors play a vital role in important medical, agricultural, and industrial applications. For example, research reactors are a vital tool in cancer therapy and radioimmunoassay blood testing. There are about 30,000 medical procedures per day in North America using medical isotopes produced in research reactors in other countries. There are also about 8,000 to 10,000 such procedures per day in Europe and a similar number on other continents. Figure S-1 provides examples of the uses and benefits of research reactors.

In the past, after irradiation in the research reactor, the used fuel (known as "spent") was transported to the United States, where it was reprocessed to extract the uranium still remaining in the spent nuclear fuel. In this way, the United States maintained complete control over the HEU that it provided to other nations. The United States began accepting HEU spent nuclear fuel from foreign research reactors in 1958.

The provision of enriched uranium from the United States to other nations was usually supported by a bilateral research agreement for each research reactor. Before 1964, these agreements provided for the lease of the enriched uranium, with explicit provision for the return of the spent nuclear fuel to the United States. After 1964, most agreements provided for the sale of this material to the foreign nation, and the United States began operating under a policy known as the "Off-Site Fuels Policy," under which the United States continued to accept, temporarily store, and reprocess the spent nuclear fuel.

What is Spent Nuclear Fuel?

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. When it is removed from a reactor, spent nuclear fuel contains some unused enriched uranium and radioactive fission products. Because of its radioactivity (primarily from gamma rays), it must be properly shielded. Nuclear fuel consists of fuel elements which can come in many configurations. Generally, a fuel element is covered by a metal called cladding and is shaped like long rods, flat plates or cylinders.

What is Enriched Uranium?

Uranium ore occurs naturally in a state that cannot be used in most reactors or to make nuclear weapons. Enriching the uranium makes it easier to use in reactors. The enrichment process increases the amount of the fissionable uranium-235 (^{235}U) isotope. Uranium enriched to contain less than 20 percent ^{235}U is called low enriched uranium. Uranium enriched to contain 20 percent or greater ^{235}U is highly-enriched uranium that can be directly used to make nuclear weapons.

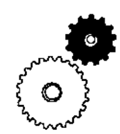
Uses and Benefits of Research Reactors

The United States has participated in cooperative international actions to expand peaceful uses of nuclear energy since the early days of the nuclear era. The foreign research reactors program has produced far-reaching benefits for medicine, science, industry, and the environment.

Advances in Nuclear Medicine

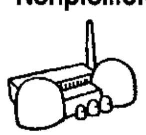
Cancer therapy, medical isotope production, clarification of the biological effects of radiation, development of improved drugs, and blood testing.

Benefits to Industry



Neutron radiography allows diagnosis of defects in metals and engines, research on new and improved materials, and leak detection.

Nonproliferation



Training of international inspectors of nuclear facilities worldwide to prevent diversion of nuclear materials.

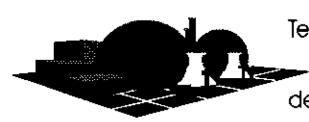
Environmental, Agricultural, and Climate Studies

Development of tracer elements for studies of pollution, waste migration, toxic waste management, mine drainage, water chemistry, sediment transport, contamination of freshwater ecosystems, atmospheric dispersion and fallout product measurements, and soil erosion.

Advancement of Basic Scientific Research

Neutron scattering experiments produce insights into elementary particle physics, clarification of the biostructure of organic substances, and development of new magnetic materials and superconducting materials.

Materials and Advanced Fuels Testing



Testing of materials and fuel forms, including safety experimentation, is being conducted to support advance fuel design and waste management development for use in the power industry.

Figure S-1 Uses and Benefits of Research Reactors

To further reduce the danger of nuclear weapons proliferation, the United States in 1978 initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program, which was aimed at reducing the use of HEU in civilian programs by promoting the conversion of foreign research reactors from HEU fuel to low enriched uranium (LEU) fuel. Research reactor fuel has become the major civilian use of HEU. As part of the RERTR program, DOE developed LEU fuel and worked with foreign research reactor operators to convert their reactors to run on such fuel.

The foreign research reactor operators who converted to LEU fuel did so in support of nuclear weapons nonproliferation objectives, even though such conversions were expensive and generally resulted in reduced capabilities of the reactors and increased operating costs. From the beginning of the RERTR program, foreign research reactor operators made it clear that their willingness to convert their research reactors to LEU fuel was contingent upon the continued acceptance by DOE of their spent nuclear fuel for disposition in the United States.

In 1986, to further encourage foreign research reactor operators to convert to LEU fuel, the DOE "Off-Site Fuels Policy" was extended to include the acceptance of spent nuclear fuel containing LEU enriched in the United States. The RERTR program has been highly successful and many foreign research reactors have been modified to operate, or have been designed to operate, with the high-density LEU fuels developed by the RERTR program. Of the 42 foreign research reactors with power levels equal to or above one million watts that use U.S. enriched fuel, 37 could operate with the currently available high-density LEU fuels. Of these, 25 are either operating on LEU fuel, or have ordered LEU fuel, and DOE anticipates that an additional eight reactors will convert to LEU fuel by 2001. Work is underway to develop improved high-density LEU fuels that would enable the remaining HEU-fueled reactors to convert as well. Thus, the RERTR program has contributed to a significant reduction in the level of HEU fuel usage in foreign research reactors.

The United States accepted foreign research reactor spent nuclear fuel until the program expired (in 1988 for HEU fuels and 1992 for LEU fuels). At that time, DOE committed to prepare an environmental review of the impacts of extending the program for accepting foreign research reactor spent nuclear fuel. In 1991, DOE issued an environmental assessment of the potential environmental impacts of the proposed extension. DOE received numerous comments from the public stating that any long-term policy should not be implemented until an EIS was prepared. DOE decided in mid-1993 to prepare an EIS to evaluate the impacts of implementing a new foreign research reactor spent nuclear fuel acceptance policy.

On April 21, 1995, DOE published a Notice of Availability (60 FR 19899) of the *Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (Draft EIS). Only spent nuclear fuel containing uranium enriched in the United States would be covered under the proposed action. The Draft EIS analyzed three Management Alternatives for implementing the proposed action: Management Alternative 1, accept and manage foreign research reactor spent nuclear fuel in the United States; Management Alternative 2, facilitate the management of foreign research reactor spent nuclear fuel overseas; and Management Alternative 3, a hybrid, or combination, of elements from the first two Management Alternatives. In Management Alternative 1, the Draft EIS assesses the impacts of managing the spent nuclear fuel at five DOE sites and using ten candidate ports of entry.

During the 90-day public comment period (April 21, 1995 to July 20, 1995), about 900 individuals attended the 17 public hearings held in or near candidate ports, management sites, and in Washington, DC. In addition to oral comments, DOE received approximately 5,040 written comments contained within approximately 1,250 comment documents on a wide range of policy, economic, and technical issues. Many commentors supported the U.S. nuclear weapons nonproliferation policy objective of seeking to reduce the use of HEU in civilian commerce. However, comments reflected a wide range of views as to which management alternative should be adopted. Some commentors supported management of the spent nuclear fuel in the United States. Other commentors questioned the need to accept spent nuclear fuel from allies and those countries that can manage their spent nuclear fuel abroad. These commentors generally believed that such spent nuclear fuel should be managed overseas. With regard to the implementation of the policy in the United States, some commentors preferred the use of military ports. Risks during transport, including those from terrorism, a sunken cask, severe shipboard fires, and the level of emergency preparedness at ports, were frequently raised as matters of concern.

In consideration of public comments, DOE has added information to the EIS, including: clarification of the proposed U.S. policy on accepting spent nuclear fuel from allies; examination of the consequences of sabotage or terrorist attack; safety of transportation casks; re-examination of the shipboard fire analysis; and general provisions of transportation and emergency response regulations and management. The Naval Weapons Station at Charleston was analyzed in addition to the other terminals of the port of Charleston that were discussed in the Draft EIS. An overview of the public comment process is presented in Section S.5 of this EIS Summary and Volume 3 of the EIS. Each public comment received is presented in Volume 3 with the DOE response. In this Final EIS, DOE and the Department of State, in consultation with other government agencies and in consideration of public comments and the EIS analysis, designated the acceptance and management of foreign research reactor spent nuclear fuel in the United States as the preferred alternative (i.e., Management Alternative 1 with modifications to several basic implementation elements).

S.1.2 Purpose and Need for Agency Action

For more than 50 years, the United States has played a leading role in international efforts to prevent the proliferation of nuclear weapons throughout the world. A key element of U.S. nuclear weapons nonproliferation policy is to reduce international commerce in HEU. DOE's and the Department of State's proposal to adopt a policy to manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States is linked to U.S. efforts to convert the foreign research reactors from HEU to LEU fuels (the latter cannot be used directly in simple nuclear weapons) and to gain worldwide acceptance of the use of LEU fuels in new research reactors.

The failure of the United States to manage foreign research reactor spent nuclear fuel could have a number of adverse consequences. Foreign governments and research reactor operators participated in the RERTR program in part because the United States accepted the spent nuclear fuel from their research reactors. The United States has not accepted HEU spent nuclear fuel for more than six years, with the exception of recent shipments of 252 spent nuclear fuel elements (153 elements from Austria, The Netherlands, Sweden, and Denmark, and 99 elements from Switzerland and Greece) under the *Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel, April 1994*. As a result, some foreign research reactor operators have run out of space to store their spent nuclear fuel and others soon will. Under such conditions, the foreign research reactor operators must either shut down their reactors, construct new storage facilities, or ship the spent nuclear fuel offsite for storage or reprocessing. Currently, overseas reprocessing results in separated HEU that is placed back into commerce for use as new reactor fuel. The overseas reprocessing facilities (e.g., Dounreay in the United Kingdom) currently do not have the special equipment to reprocess the high-density LEU fuels that the United States is encouraging foreign research reactors to use to replace the HEU fuels. Thus, in the absence of action to resolve the question of the disposition of spent nuclear fuel, any foreign research reactor operator who reprocesses spent nuclear fuel to control a spent fuel inventory must continue to use, or convert back to, fuel containing HEU. Some nations, such as Belgium and Germany, have already begun shipments for reprocessing. For most foreign research reactor operators, construction of a new storage facility would not be practical due to the very high cost of storing small amounts of spent nuclear fuel and the long time required to design, license, and construct facilities. The most realistic near-term option for these reactor operators (particularly those in countries without power reactor programs) is to ship their spent nuclear fuel offsite for reprocessing. In such a case, foreign research reactor operators would have little incentive to convert their reactors to LEU fuels.

A crucial consideration in making the proposal to manage foreign research reactor spent nuclear fuel was the then upcoming 1995 international conference on the *Treaty on the Non-Proliferation of Nuclear Weapons*. At that conference, a major United States foreign policy objective was reached when the parties agreed by consensus to make the Treaty a permanent part of the international nuclear nonproliferation regime. One key to the success of the conference was the ability of the United States to convince other Treaty parties that the nuclear weapons States had complied with their obligations

The Treaty on the Non-Proliferation of Nuclear Weapons

The 1968 *Treaty on the Non-Proliferation of Nuclear Weapons* is the basis for the world's nuclear weapons nonproliferation regime. The purpose of the Treaty is to keep the number of countries with nuclear weapons to the five countries that possessed such weapons before 1967: the United States, Russia, the United Kingdom, France, and China. In addition to the five nuclear weapons States, 175 other countries are members of the Treaty. On May 12, 1995, the Review and Extension Conference of the Parties to the Treaty agreed by consensus to extend the Treaty for an indefinite period. This accomplishment achieved a major goal of United States foreign policy. The obligations for compliance with the *Treaty on the Non-Proliferation of Nuclear Weapons* apply to both nuclear weapons States and nonnuclear weapons States. While nonnuclear weapons States agree not to pursue development or acquisition of nuclear weapons or other nuclear explosive devices, the nuclear weapons States commit themselves to work toward the ultimate elimination of their nuclear arsenals. All States are thus bound to help reduce the global threat of nuclear weapons, but must do so without prejudice to a nation's ability to pursue the benefits of peaceful uses of nuclear energy.

under Article IV of the Treaty and had shared with nonnuclear weapons States the benefits of peaceful nuclear cooperation.

The parties also agreed to review the Treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors, or has been forced to seek reprocessing, could accuse the United States of not having complied with its Treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to U.S. interests.

To illustrate the level of concern that exists, DOE has received letters from the U.S. Department of State, the Nuclear Regulatory Commission (NRC), the Arms Control and Disarmament Agency, and the International Atomic Energy Agency, all urging DOE to implement a new policy to manage the foreign research reactor spent nuclear fuel. (See Appendix G of the Final EIS.)

By proposing a policy for management of certain foreign research reactor spent nuclear fuel, DOE and the Department of State do not seek to indefinitely accept or otherwise manage spent nuclear fuel from foreign research reactors. Rather, the purpose of the proposed new policy is to remove as much U.S.-origin HEU as possible from international commerce while giving the foreign research reactor operators and their host countries time to convert to operation with LEU fuel and to make their own arrangements for disposition of subsequently generated LEU spent nuclear fuel. Should the proposed policy be adopted, the foreign research reactor operators and countries in which the research reactors are operating must be prepared to implement their own arrangements for disposition of their spent nuclear fuel after the policy expires.

S.1.3 Decisions to be Made Based on this EIS

The principal policy decision for which this EIS will provide a basis is whether the United States should adopt a policy for the management of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. The countries which host foreign research reactors covered under this EIS are identified in Figure S-2.

Should a decision be made to manage this foreign research reactor spent nuclear fuel in the United States, decisions also would have to be made on the duration of the policy, amount of fuel to be accepted, transportation modes, ports of entry, and method of spent nuclear fuel management (storage, chemical separation, or use of a new treatment and/or packaging technology). Should the decision be made to facilitate management of foreign research reactor spent nuclear fuel overseas, decisions would need to be made on what assistance the United States would provide to foreign nations for storage or reprocessing of the spent nuclear fuel overseas. The decisions of DOE and the Department of State will be announced in the Record of Decision for this EIS, which will be available no less than 30 days after the Environmental Protection Agency publishes a Notice of Availability for the Final EIS.

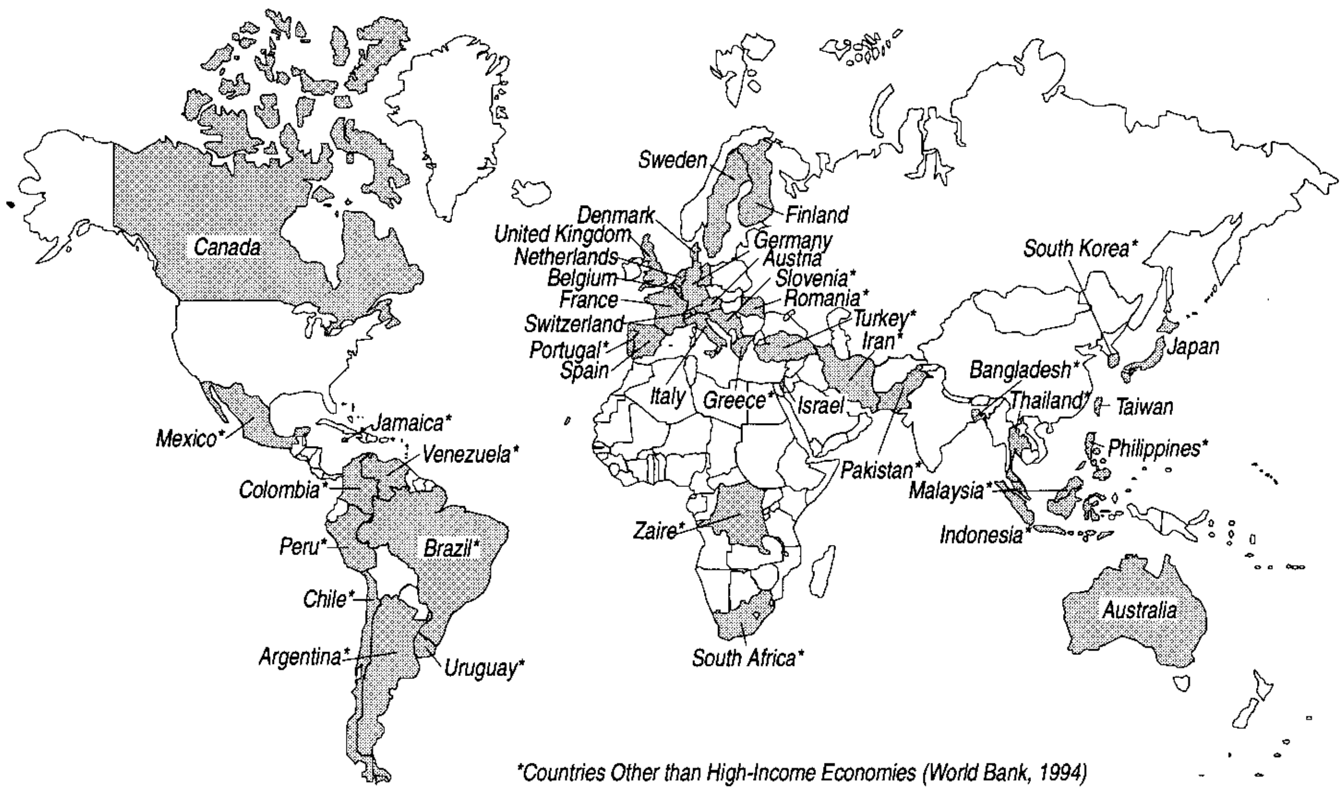


Figure S-2 Countries which Host the Research Reactors

S.1.4 Relationship of This EIS to Other NEPA Documentation and Reports Relating to Spent Nuclear Fuel Management

Certain potential actions discussed in this EIS would depend on decisions to be made under other NEPA analyses. For example, the site(s) at which foreign research reactor spent nuclear fuel would be managed (if the spent nuclear fuel were to be accepted in the United States) were considered in Volume 1 of the *DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, or "Programmatic SNF&INEL Final EIS," issued in April 1995. The five management sites considered were: the Savannah River Site, the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, and the Nevada Test Site. The Record of Decision, issued on May 30, 1995, indicated that DOE aluminum clad spent nuclear fuel will be managed at the Savannah River Site and other DOE spent nuclear fuel will be managed at the Idaho National Engineering Laboratory. Accordingly, the Comment Response Document (Volume 3) for this EIS focuses on the Savannah River Site and the Idaho National Engineering Laboratory, although to maintain maximum consistency with the analysis provided in the Programmatic SNF&INEL Final EIS, this EIS analyzes the impacts of the proposed action at all five sites.

Potential chemical separation activities for nuclear materials already in inventory at the Savannah River Site are addressed in the *Interim Management of Nuclear Materials Final Environmental Impact Statement*. A Record of Decision and Notice of Preferred Alternative was published in December 1995 in the *Federal Register* (60 FR 65300). Decisions were made in the Record of Decision for the majority of materials covered by the EIS and processing Mark-16 and Mark-22 fuels and blending down the resulting HEU to LEU was identified as the preferred alternative. These fuels are similar to the aluminum-based foreign research reactor spent nuclear fuel, although significant corrosion has been identified. An amended Record of Decision is expected soon regarding the Mark-16 and Mark-22 spent nuclear fuel. DOE has taken into consideration the Record of Decision on the *Interim Management of Nuclear Materials Final EIS* in preparation of this EIS and in reaching a decision on how to implement the proposed policy, if adopted.

The relationship of this EIS to other DOE NEPA reviews, either completed or currently under preparation, and other DOE analyses related to the EIS, is discussed in Volume 1, Section 1.5 of the EIS.

S.2 Proposed Action and Alternatives

The proposed action is for DOE and the Department of State to jointly adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed action. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and eventually eliminate, HEU from civilian commerce. The proposed policy applies solely to aluminum-based and Training, Research, Isotope, General Atomic (TRIGA) foreign research reactor spent fuels and target material containing HEU and LEU of U.S. origin.

To implement the proposed action, the EIS analyzes three "Management Alternatives," which are:

Management Alternative 1: Accept and manage foreign research reactor spent nuclear fuel in the United States. This could be implemented by accepting foreign research reactor spent nuclear fuel (containing HEU or LEU enriched in the United States) for management in the United States.

Management Alternative 2: Facilitate the management of foreign research reactor spent nuclear fuel overseas. This could be implemented by U.S. assistance in spent nuclear fuel storage or reprocessing.

***Management Alternative 3*:** A hybrid, or combination, of elements from the above two Management Alternatives.

Each management alternative has further implementation components and alternatives, as identified in Figure S-3. These are addressed in succeeding sections.

The EIS also evaluates the "No Action" alternative, in which case the United States would take no action concerning such a policy.

DOE did not identify a preferred alternative for the management of foreign research reactor spent nuclear fuel in the Draft EIS. After careful consideration of public comments on the Draft EIS and other factors, DOE and the Department of State have designated Management Alternative 1, with modifications to several basic implementation

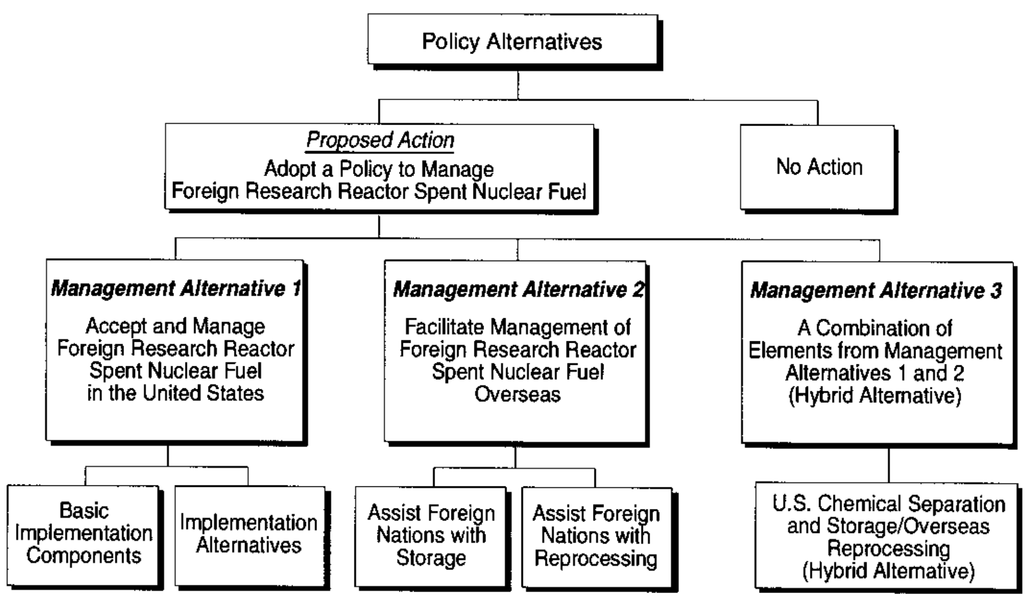


Figure S-3 Management Alternatives of the Proposed Action elements, as the preferred alternative for the implementation of the proposed policy. This preferred alternative is to accept and manage in the United States up to 22,700 elements of foreign research reactor spent nuclear fuel containing uranium enriched in the United States and target material. The preferred alternative is described in Section S.2.3 of this Summary.

S.2.1 Overview of Management Alternatives to Implement the Proposed Action

The three Management Alternatives are summarized below.

Management Alternative 1: Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

Under Management Alternative 1, foreign research reactor spent nuclear fuel, which contains uranium enriched in the United States, would be transported to the United States in casks designed on the basis of international regulations that are essentially identical to those promulgated by the NRC and certified by the U.S. Department of Transportation. In accordance with the Record of Decision for the Programmatic SNF&INEL Final EIS, all of the aluminum clad foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel, such as the TRIGA elements, to be accepted by DOE would be managed at the Idaho National Engineering Laboratory, pending ultimate disposition. Nevertheless, all five of the spent nuclear fuel management sites originally considered in the Draft EIS have been kept in this Final EIS to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL Final EIS. The components of the basic implementation of Management Alternative 1 are identified in Figure S-4.

The EIS also evaluates several different options for implementing Management Alternative 1. (Indeed, the preferred alternative incorporates a combination of various implementation alternatives that were analyzed.) The implementation alternatives are identified in Figure S-5. They include, for example, different time periods for the policy duration, different storage technologies, and a chemical separation alternative to storing the fuel.

Management Alternative 2: Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

This Management Alternative would require bilateral agreements between the United States and one or more foreign governments in order to ensure consistency with U.S. nuclear weapons nonproliferation policy. Under this Management Alternative there are two subalternatives: one is to provide assistance to foreign nations that are able to store their spent nuclear fuel in facilities in their own countries, and a second is to provide nontechnical (financial and/or logistical) assistance in reprocessing the spent nuclear fuel overseas in facilities operated under international safeguards sufficient to satisfy U.S. nuclear weapons nonproliferation concerns.

Under the first subalternative, DOE and the Department of State would provide assistance, incentives, and coordination for storage at one or more locations overseas, with appropriate storage technologies, regulations, and safeguards. In the second

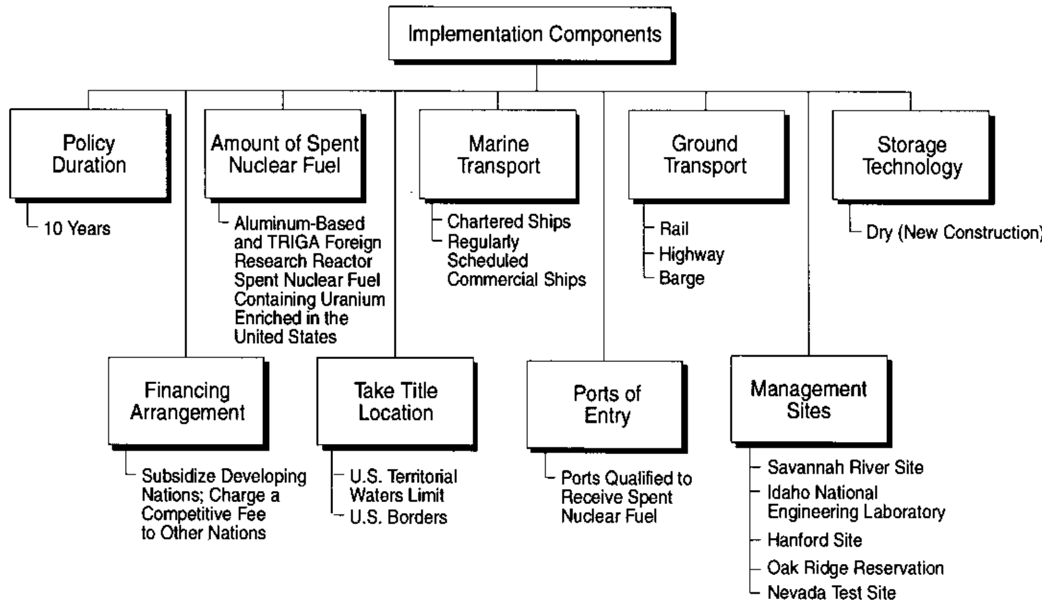


Figure S-4 Basic Implementation Components of Management Alternative 1

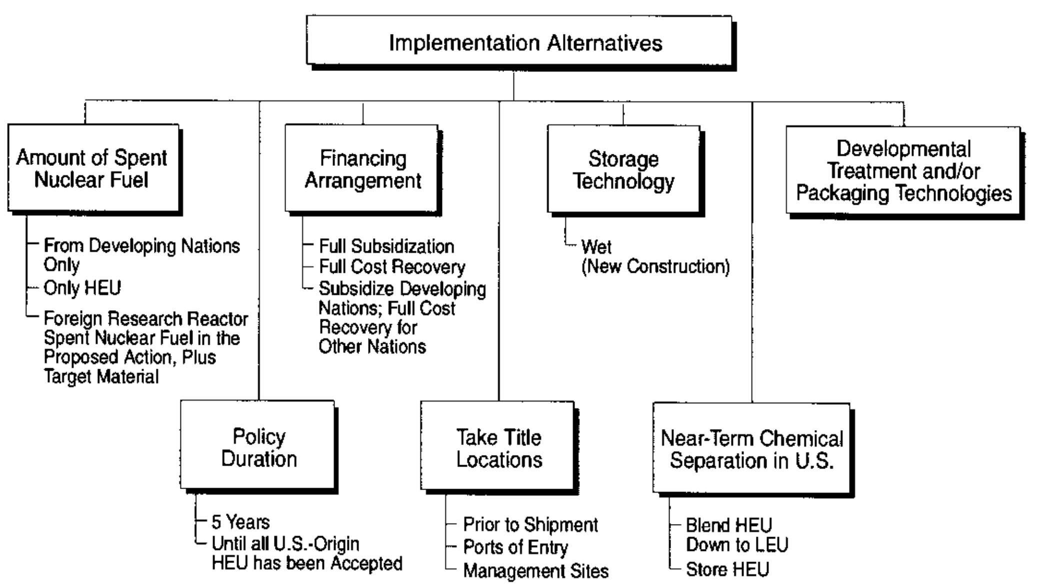


Figure S-5 Implementation Alternatives subalternative, DOE and the Department of State would provide nontechnical assistance, incentives, and coordination to foreign research reactor operators and reprocessors to facilitate reprocessing of spent nuclear fuel overseas in facilities operated under international inspections and safeguards. Facilities operated by the United Kingdom Atomic Energy Authority at Dounreay, United Kingdom, and by Cogema at Marcoule, France might be used for this purpose. After reprocessing, the recovered HEU would be blended down to LEU at these same facilities for reuse as either LEU research reactor fuel or commercial power reactor fuel. The high-level wastes resulting from this reprocessing would be sent to the country in which the spent nuclear fuel was irradiated. If the reprocessing wastes could not be sent to the country in which the spent nuclear fuel was irradiated, such wastes would be accepted by the United States for storage and ultimate geologic disposal.

Management Alternative 3: A Combination of Elements From Management Alternatives 1 and 2 (Hybrid Alternative)

Under Management Alternative 3, DOE and the Department of State would combine elements from Management Alternatives 1 and 2 to develop new alternatives for management of foreign research reactor spent nuclear fuel in the United States or abroad. For example, DOE and the Department of State could combine partial storage or reprocessing overseas with partial storage or chemical separation in the United States.

The following sections discuss in more detail the implementation of each Management Alternative.

S.2.2 Management Alternative 1 - Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

This section provides a more detailed summary of Management Alternative 1 and identifies components of its basic implementation and components of various implementation alternatives.

S.2.2.1 Basic Implementation Components

The components of the basic implementation of Management Alternative 1 (see Figure S-4) provide the foundation for the analyses of impacts presented in the EIS. They are:

- Policy Duration
- Financing Arrangement
- Amount of Foreign Research Reactor Spent Nuclear Fuel
- Location for Taking Title to Foreign Research Reactor Spent Nuclear Fuel
- Marine Transport
- Port(s) of Entry
- Ground Transport

- Foreign Research Reactor Spent Nuclear Fuel Management Sites
- Storage Technologies.

S.2.2.1.1 Policy Duration

The policy duration would be the 10-year period beginning on the date when the policy takes effect. Spent nuclear fuel containing HEU and LEU of U.S. origin that is currently being stored or is to be generated during the 10-year policy period would be accepted. Actual shipments of spent nuclear fuel to the United States could be made for a period of 13 years starting from the effective date of the policy implementation, as long as spent nuclear fuel was generated within the 10-year policy period. The additional three years would allow for a cooling time for fuel discharged from a reactor late in the policy period, logistics in arranging for shipment of this fuel, and other unplanned for delays.

S.2.2.1.2 Financing Arrangement

The United States would bear the full cost of transporting and managing the foreign research reactor spent nuclear fuel received from countries with other-than-high-income-economies. For high-income economy countries, the United States would charge a competitive fee for all spent nuclear fuel management activities conducted by the United States.

S.2.2.1.3 Amount of Foreign Research Reactor Spent Nuclear Fuel

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation of Management Alternative 1 is up to about 19.2 MTHM from up to approximately 22,700 individual spent nuclear fuel elements (1 MTHM equals about 2,200 pounds).

S.2.2.1.4 Location for Taking Title to Foreign Research Reactor Spent Nuclear Fuel

DOE would take title to the foreign research reactor spent nuclear fuel when the fuel entered U.S. territorial waters (19 km or 12 miles offshore) or crossed U.S. continental borders for shipments from Canada.

As a comparison:

- *During the last 5 decades, DOE and its predecessor agencies have produced, transported, received, stored, and processed more than 100,000 metric tons of heavy metal (MTHM) of spent nuclear fuel.*
- *Currently about 2,700 MTHM of DOE spent nuclear fuel are being stored at various DOE facilities.*
- *Currently, about 30,000 MTHM of spent nuclear fuel from commercial reactors are stored at reactor sites in the United States.*

S.2.2.1.5 Marine Transport

DOE estimates that 721 cask loads of foreign research reactor spent nuclear fuel would be sent to the United States by ship over a 13-year acceptance period under the basic implementation of Management Alternative 1.

S.2.2.1.6 *Port(s) of Entry*

The receipt of the foreign research reactor spent nuclear fuel could occur at any of the following candidate ports of entry:

- Charleston, SC (includes Naval Weapons Station and Wando Terminal, Mt. Pleasant)
- Galveston, TX
- Hampton Roads, VA (includes Terminals at Newport News, Norfolk, and Portsmouth, VA)
- Jacksonville, FL
- Military Ocean Terminal Sunny Point, NC
- Naval Weapons Station Concord, CA
- Portland, OR
- Savannah, GA
- Tacoma, WA
- Wilmington, NC

The locations of these ports in relation to the five candidate management sites are depicted on the map in Figure S-6.

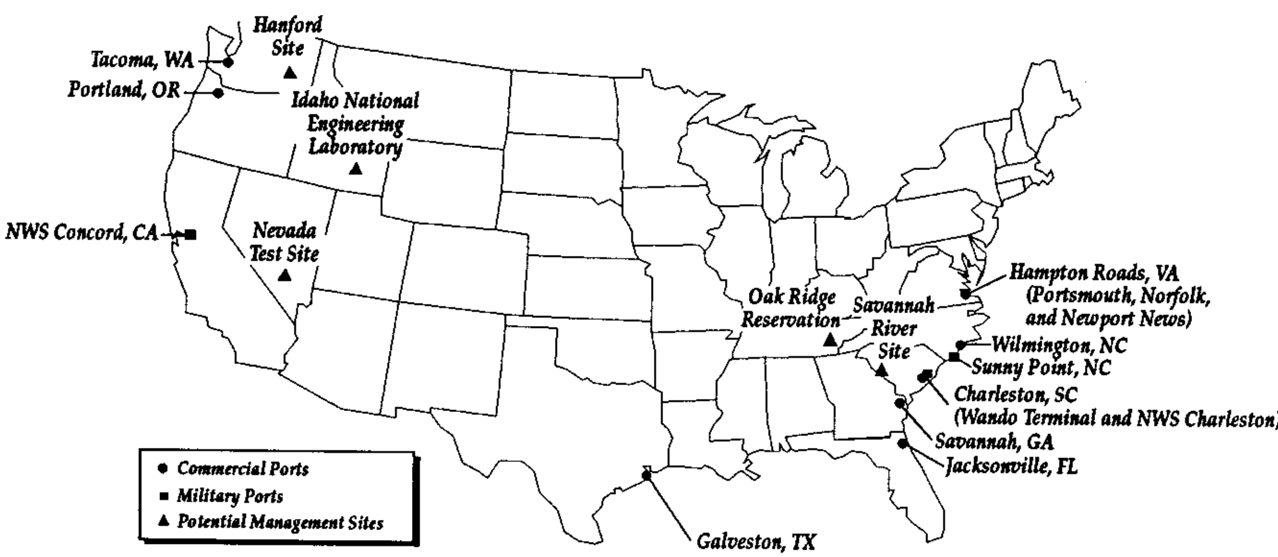


Figure S-6 Location of Potential Ports of Entry and Management Sites

The potential ports of entry were identified using screening criteria that included appropriate experience, safe transit, adequate facilities, and population around the ports and along routes to potential management sites. Screening criteria were based on input from the public (during the EIS scoping process), a U.S. Merchant Marine Academy panel of maritime safety experts, and factors identified in Section 3151 of the National Defense Authorization Act for Fiscal Year 1994.

S.2.2.1.7 Ground Transport

The basic implementation of Management Alternative 1 would involve transporting casks containing foreign research reactor spent nuclear fuel by truck, rail, or barge from the ports of entry or Canadian border crossings to potential management sites. It could also involve later transport of the spent nuclear fuel between the management sites.

All spent nuclear fuel shipments must comply with both NRC and Department of Transportation regulatory requirements. Specific highway routing of the cask shipments would follow a systematic process in accordance with Department of Transportation regulations. Shipments must also comply with NRC regulations covering physical security and notification.

Rail routing is not covered by specific Department of Transportation and NRC regulations. Therefore, shippers would generally select the most direct available rail route, which would serve to reduce travel time and radiation exposure consistent with track class and other rail service requirements.

S.2.2.1.8 Foreign Research Reactor Spent Nuclear Fuel Management Sites

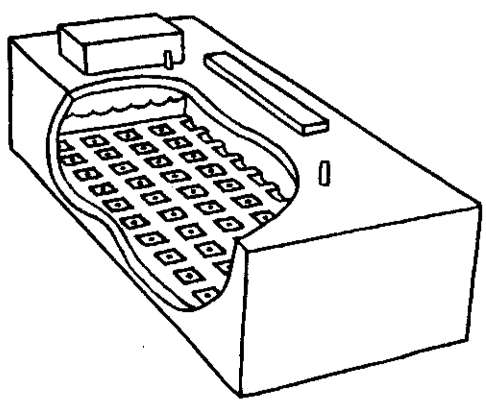
Potential sites for the receipt and management of foreign research reactor spent nuclear fuel have been specified by DOE in the Record of Decision for the Programmatic SNF&INEL Final EIS, which is concerned with the environmental impacts of management of spent nuclear fuel. In accordance with this Record of Decision, all of the aluminum clad foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel, such as the TRIGA elements, to be accepted by DOE would be managed at the Idaho National Engineering Laboratory, pending ultimate disposition. Notwithstanding the Record of Decision of the Programmatic SNF&INEL Final EIS, full analyses of all five sites are included in the EIS to maintain analytical consistency with the programmatic analyses.

In the analyses considering use of potential management sites other than the Savannah River Site and the Idaho National Engineering Laboratory, the near-term unavailability of the other three candidate management sites to accept foreign research reactor spent nuclear fuel at the beginning of the implementation period (due to lack of existing storage capacity) would necessitate temporary receipt and storage of the spent nuclear fuel at either the Savannah River Site or the Idaho National Engineering Laboratory. The other three sites — the Oak Ridge Reservation, the Hanford Site, and the Nevada Test Site — would not have facilities available for approximately 10 years. The Nevada Test Site could receive the spent nuclear fuel in approximately five years if a decision were made to refurbish the Engine Maintenance and Disassembly (E-MAD) facility rather than construct a new facility.

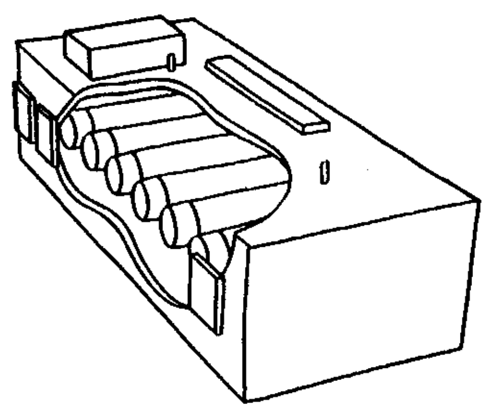
S.2.2.1.9 Storage Technologies

Under the basic implementation of Management Alternative 1, DOE would manage foreign research reactor spent nuclear fuel for a period starting in 1996, and continuing for 40 years, or until ultimate disposition. The technology for safely storing spent nuclear fuel has been in use for over 40 years in the nuclear industry. Spent nuclear fuel storage is generally characterized as either "wet" or "dry." Wet storage means that the spent nuclear fuel elements reside in a water-filled pool. Dry storage means that the fuel is stored in a dry enclosed atmosphere. During the first few years, storage would take place in existing storage facilities that use both wet and dry storage technologies. For the period beyond those first few years, when construction of new facilities may become necessary, the storage technology evaluated under the basic implementation is dry storage. However, construction of new wet storage facilities is considered as an implementation alternative. Figure S-7 depicts typical wet and dry storage facilities.

The wet pool type of spent nuclear fuel storage is used at almost every water-cooled nuclear reactor in the world. There are currently more than 600 operating water-cooled power and research nuclear reactors, each with an individual storage pool. Dry storage technology involves the encapsulation of spent nuclear fuel in a steel cylinder that may be placed in a concrete or massive steel cask or structure. Different forms of dry spent nuclear fuel storage have been used for over 40 years in the nuclear industry. Whether wet or dry storage is used, the facilities are designed or have been upgraded to withstand natural phenomena such as earthquakes, floods, tornadoes, hurricanes, high and low temperatures, and wind generated missiles (branches, poles, etc.). The designs also include provisions to mitigate sabotage or terrorist acts.



Typical Wet Storage Facility



Typical Dry Storage Facility

Figure S-7 Typical Storage Facilities

S.2.2.1.10 Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such spent nuclear fuel). However, since the repository characterization program is in its early stages, the waste acceptance criteria for disposal of DOE's spent nuclear fuel in a repository have not been developed. Thus, a determination cannot be made at this time as to the requirements that must be met to allow emplacement of the foreign research reactor spent nuclear fuel in the repository. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, disposal of vitrified high-level waste resulting from chemical separation, as well as utilization of various potential new technologies to process the spent nuclear fuel into a more stable form prior to its ultimate disposition. In the event that the availability of a geologic repository were to be delayed beyond the 40-year program period, DOE assumed for purposes of this analysis that it would continue to manage the foreign research reactor spent nuclear fuel, or the high-level radioactive waste resulting from the chemical separation or other processing of such spent nuclear fuel, at the management sites until a geologic repository becomes available. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under NEPA.

S.2.2.2 Implementation Alternatives for Management Alternative 1

This EIS also evaluates a range of implementation alternatives that modify one of the basic implementation components of Management Alternative 1 (see Figure S-5). The implementation alternatives (and implementation subalternatives) include the following:

1. Alternative amount of foreign research reactor spent nuclear fuel to be accepted:
   a. Only from countries with other-than-high-income-economies (up to 1.9 MTHM; 5,000 elements)
   b. HEU only (up to 4.6 MTHM; 11,200 elements)
   c. Target material in addition to spent nuclear fuel (up to 0.6 MTHM; equivalent to 620 elements)

2. Alternative policy durations:
   a. Five-year policy (up to 13 MTHM; 18,800 elements)
   b. Indefinite HEU/10-year LEU policy (same amount as basic implementation; different timing)

3. Alternative financing arrangements:
   a. Subsidize all countries

b. Charge all countries full cost of accepting and managing foreign research reactor spent nuclear fuel
c. Subsidize other-than-high-income economy countries; charge high-income economy countries full-cost recovery fee

4. Alternative locations for taking title:

   a. Prior to shipment
   b. Port(s) of entry
   c. Management sites

5. Wet storage technology for new construction

6. Near-term conventional chemical separation in the United States[1]

   a. Extent of chemical separation: dedicated to foreign research reactor spent nuclear fuel only, or part of larger-scale DOE chemical separation activities
   b. Uranium disposition: blend HEU down to LEU or process HEU to oxide for interim storage

7. Developmental treatment and/or packaging technologies (Conduct a development program leading to a decision on whether to construct and operate a cost-effective new treatment and/or packaging facility. The objective of this technical strategy is to treat, package, and store spent nuclear fuel in a manner suitable for placement into a geologic repository.

S.2.3 Preferred Alternative

In selecting a preferred alternative for the management of foreign research reactor spent nuclear fuel, DOE and the Department of State took several factors into consideration, including the following:

1. U.S. Government nuclear weapons nonproliferation policies and objectives;

2. DOE responsibilities (e.g., safe handling of hazardous materials, safety/health risks to workers, compatibility with other ongoing missions, etc.);

3. Potential environmental impacts (e.g., public safety, etc.);

4. Public comments received and concerns expressed following issuance of the Draft EIS;

5. Analysis of impacts and alternatives in the Programmatic SNF&INEL Final EIS (DOE, 1995c), as well as the Record of Decision for that EIS;

---

1 *Chemical separation of foreign research reactor spent nuclear fuel in existing facilities is not preferred by DOE as a technology for routine management of spent nuclear fuel in the United States because of the additional waste streams that would be generated when these activities are conducted. Nonetheless, chemical separation remains a reasonable alternative in light of DOE's substantial technical expertise in these operations and the availability of existing facilities.*

6. Estimated costs of alternatives for management of foreign research reactor spent nuclear fuel;

7. Public issues/concerns/perceptions (e.g., fairness/equity to affected States and populations, etc.); and

8. Uncertainties (e.g., future budget priorities and continuity of funding, technology development, repository timing and waste form acceptance criteria, regulatory change, etc.).

Based on consideration of these factors, DOE and the Department of State, in consultation with other Government agencies, designate the alternative described below as the preferred alternative. This preferred alternative is the same as Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States, discussed in Section 2.2 of the EIS and S.2.2 of the Summary), with the modifications discussed below. The basic components of Management Alternative 1 have been modified to incorporate various implementation alternatives discussed in Section 2.2.2 of the EIS and S.2.2.2 of the Summary.

The amount of foreign research reactor spent nuclear fuel that would be accepted and managed, as specified in Section 2.2.1.3 of the EIS, could total approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The target material that would be accepted and managed, as specified in Section 2.2.2.1 of the EIS, contains an additional 0.6 MTHM representing the uranium content of approximately 620 additional typical foreign research reactor spent nuclear fuel elements. The following stipulations on qualifying spent nuclear fuel types would apply:

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy

Preferred Alternative Elements

Policy: Adopt a policy to accept and manage foreign research reactor spent nuclear fuel and target material in the United States.

Amount of Fuel to be Accepted: Up to 19.2 metric tons of heavy metal in 22,700 fuel elements, and 0.6 metric tons of heavy metal of target material.

Policy Duration: Ten years. Shipment to United States could occur for 13 years.

Financing Arrangements: United States would bear the full cost for transporting and managing the spent nuclear fuel accepted from countries with other-than-high-income economies, and would charge high-income economy countries a fee.

Marine Transport: Either chartered or commercial ships.

Ports of Entry: Military ports of Charleston Naval Weapons Station, SC, and Naval Weapons Station Concord, CA.

Location for Taking Title: Upon unloading the spent nuclear fuel at U.S. ports of entry and at the U.S.-Canadian border.

Ground Transport: Truck or rail.

Management Sites: Aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site. TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

Management Technologies: Management of the TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be based on the use of existing storage facilities with the possible use of a new treatment and/or packaging technology.

Management of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site would be based on the use of existing storage facilities, development and implementation of a new treatment and/or packaging technology, and chemical separation if necessary.

DOE would conduct an independent study of the nonproliferation and other implications of reprocessing a portion of the foreign research reactor spent nuclear fuel at F-Canyon prior to committing to the use of reprocessing for other than health or safety reasons.

becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.

- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.

- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.

- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).

- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

The policy duration under this preferred alternative would be 10 years, beginning on the date when the management policy would become effective, as discussed in Section 2.2.1.1 of the EIS. Shipments of spent nuclear fuel to the United States could be made for a period of 13 years, starting from the effective date of policy implementation, as long as the spent nuclear fuel had already been discharged prior to the beginning of the policy period or is discharged during the policy period.

The aluminum-based foreign research reactor spent nuclear fuel would be managed at the Savannah River Site and the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory, in accordance with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the settlement agreement reached between DOE and the State of Idaho [Public Service Co. of Colorado v. Batt, No. CV 91-0035-S-EJL (D. Id.) and United States v. Batt, No. CV-91-0054-S-EJL (D. Id.)]. Under this preferred alternative, up to approximately 19 MTHM of aluminum-based foreign research reactor spent nuclear fuel (approximately 17,800 elements), representing up to approximately 675 casks, and target material representing up to approximately 140 additional casks would be accepted and managed at the Savannah River Site. Also, up to approximately 1 MTHM of TRIGA foreign research reactor spent nuclear fuel (approximately 4,900 elements), representing up to approximately 162 casks would be accepted and managed at the Idaho National Engineering Laboratory.

The candidate U.S. ports of entry are listed in Section 2.2.1.6 of the EIS and S.2.2.1.6 of the Summary, and are described in detail in Chapter 3 of the EIS. Although all of the candidate ports are acceptable based on the port selection criteria discussed in Appendix D, DOE would prefer to use the military ports in proximity to the spent nuclear fuel management sites (i.e., Charleston Naval Weapons Station and the Naval Weapons

Station Concord). Under this preferred alternative, a maximum of 38 casks of TRIGA foreign research reactor spent nuclear fuel (estimated to require about 5 shipments) could be accepted at a western port, with 150 to 300 shipments being accepted via an eastern port.

The foreign research reactor spent nuclear fuel and target material would be shipped by either chartered or regularly scheduled commercial ships from the foreign ports to the United States, as specified in Section 2.2.1.5 of the EIS.

DOE would take title to the foreign research reactor spent nuclear fuel and target material that is shipped by sea after it is offloaded at the port of entry, and to the spent nuclear fuel and target material shipped solely overland (i.e., from Canada) at the border crossing between Canada and the United States.

The foreign research reactor spent nuclear fuel and target material would be transported from the United States ports to the management sites by truck and rail as specified in Section 2.2.1.7 and S.2.2.1.7 of the Summary.

The financing arrangement under this preferred alternative would be for the United States to bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel and target material accepted from countries with other-than-high-income economies, and to charge high-income economy countries a competitive fee. The fee would be established in a *Federal Register* Notice (as opposed to being published in this Final EIS), to allow DOE flexibility to adjust the fee to account for inflation, or changes in spent nuclear fuel management practices in the United States.

For the aluminum-based foreign research reactor spent nuclear fuel, a three point strategy is proposed, as follows:

1. DOE would embark immediately on an accelerated program at the Savannah River Site to identify, develop, and demonstrate one or more non-reprocessing, cost-effective treatment and/or packaging technologies to address potential health and safety issues that may develop and to prepare the foreign research reactor spent nuclear fuel for ultimate disposal. The purpose of any new facilities that might be constructed to implement these technologies would be to change the foreign

Developmental Treatment and/or Packaging Technology Options for Spent Nuclear Fuel

Direct Disposal in Small Packages: Place fuel into small waste packages with neutron poison to control criticality.

Dissolve and Vitrify: Dissolve and mix fuel with depleted uranium to produce LEU and vitrify the mixture.

Melt and Dilute/Poison: Melt and dilute or mix fuel with a neutron poison.

Chop and Dilute/Poison: Chop fuel and dilute with depleted uranium or mix with a neutron poison.

Plasma Arc Treatment: Place fuel into plasma centrifugal furnace with other material to melt and convert into a ceramic material.

Electrometallurgical Treatment: Melt fuel in an electrolytic cell to remove the bulk of the aluminum (for disposal as low-level waste); vitrify the residual aluminum, actinides and fission products; recover pure uranium if required.

Glass Material Oxidation and Dissolution System: Melt fuel with glass-forming-materials in a glass melt furnace to form glass.

Can-in-Canister: Place fuel, with a critically safe quantity of uranium, in a can and place that can into a canister and surround with high-level waste glass from the Defense Waste Processing Facility.

Chloride Volatility: React fuel with chlorine gas to convert all materials into a volatile gas. Separate uranium, actinides, and fission products by cooling and distillation.

research reactor spent nuclear fuel into a form that is suitable for geologic disposal, without necessarily separating the fissile materials, while meeting or exceeding all applicable safety and environmental requirements. Examples of technologies that would be considered include: *can-in-canister, chop and dilute/poison, melt and dilute/poison, plasma arc treatment, electrometallurgical treatment, glass material oxidation and dissolution, chloride volatility, dissolve and vitrify, direct disposal in small packages*, etc. In conjunction with the examination of new technologies, variations of conventional direct disposal methods would also be explored. After treatment and/or packaging, the foreign research reactor spent nuclear fuel would be managed on site in "road ready" dry storage until transported off-site for continued storage or disposal. DOE would select, develop, and implement, if possible, one or more of these treatment and/or packaging technologies by the year 2000. DOE is committed to avoiding indefinite storage of this spent nuclear fuel in a form that is unsuitable for disposal.

2. Despite DOE's best efforts, it is possible that a new treatment and/or packaging technology may not be ready for implementation by the year 2000. It may become necessary, therefore, for DOE to use the F-Canyon to reprocess some foreign research reactor spent nuclear fuel elements, while the F-Canyon is operating to stabilize at-risk materials as recommended by the Defense Nuclear Facilities Safety Board. (For example, under current schedules this activity could take place between the years 2000 and 2002.) In that event, the foreign research reactor spent nuclear fuel would be converted into LEU and wastes generated during reprocessing. Certain wastes would be vitrified in the Defense Waste Processing Facility, while others would be solidified in the Saltstone facility. In order to provide a sound policy basis for making a determination on whether and how to utilize the F-Canyon for processing tasks that are not driven by health and safety considerations, DOE will commission or conduct an independent study of the nonproliferation and other (e.g., cost and timing) implications of reprocessing spent nuclear fuel from foreign research reactors. The study will be initiated in mid-1996 and will be completed in a timely fashion to allow a subsequent decision about possible use of the F-Canyon for foreign research reactor spent nuclear fuel reprocessing to be fully considered by the public, the Congress and the Executive Branch agencies. Pending disposition of the foreign research reactor spent nuclear fuel by either a new treatment and/or packaging technology or reprocessing in the F-Canyon, the spent nuclear fuel would be placed in existing wet storage at the Savannah River Site.

3. DOE would conduct a program of close monitoring of any foreign research reactor spent nuclear fuel and target material that would be accepted for storage in existing wet storage facilities. DOE is presently unaware of any technical basis for believing that this spent nuclear fuel cannot be safely stored until one or more of the treatment and/or packaging technologies becomes available. Nevertheless, if health and safety concerns involving any of the foreign research reactor spent nuclear fuel elements are identified prior to development of an appropriate treatment and/or packaging technology, DOE would use the F-Canyon to reprocess the affected spent nuclear fuel elements, if it is still operating to stabilize at-risk materials.

Because of criticality constraints stemming from the configuration of the F-Canyon, under no circumstances would it be possible to produce separated HEU that is suitable for a nuclear weapon. Instead, depleted uranium would be added to the foreign research reactor spent nuclear fuel near the beginning of the reprocessing process, so that only LEU would be produced when the uranium is separated from the fission products. The trace quantities of plutonium in the spent nuclear fuel would be left in and solidified along with the high-level radioactive reprocessing wastes. This would further the President's policy to discourage the accumulation of excess weapons-grade fissile materials, to strengthen controls and constraints on these materials and, over time, to reduce worldwide stocks.

The TRIGA foreign research reactor spent nuclear fuel would be stored at the Idaho National Engineering Laboratory in the Fluorinel Dissolution and Fuel Storage (FAST) facility (wet storage) or preferably the dry storage Irradiated Fuel Storage Facility (IFSF) and the CPP-749 dry storage area. After 2003, all foreign research reactor spent nuclear fuel would be managed in accordance with the provisions of the settlement agreement between DOE and the State of Idaho, until transported off-site for ultimate disposition. Depending on the nature of any new treatment and/or packaging technology that might be developed, the TRIGA spent nuclear fuel would also be processed using such a new technology, if necessary for disposal.

A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, whose goal is minimizing and eventually eliminating the use of HEU in civil nuclear programs, by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, are dependent on the United States' commitment to action such as that embodied in this preferred alternative.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations and persons as a signal of endorsement of the use of reprocessing as a routine method of waste management for spent nuclear fuel or a reversal of U.S. policy on reprocessing. This would not be an accurate interpretation. The U.S. policy regarding reprocessing was established in Presidential Decision Directive 13. DOE and the Department of State have determined that this preferred alternative is not inconsistent with that policy. The draft version of this EIS indicated that reprocessing is a non-preferred technology and would not be used unless one or more of a set of specific conditions occurred (Section 2.2.2.6 of the Draft EIS). This final preferred alternative, which includes reprocessing, establishes a prescribed set of circumstances that would have to be met before reprocessing would be used. The independent study discussed above in point 2 of the strategy for management of aluminum-based spent nuclear fuel will review the policy, technology, cost and schedule implications for reprocessing foreign research reactor spent nuclear fuel to determine whether reprocessing of foreign research reactor spent nuclear fuel is justified for other than health and safety reasons.

Policy considerations and environmental impacts associated with implementation of this preferred alternative are presented in Section 4.7 of the EIS and S.4.4.1 and S.4.4.2 of the Summary. Cost considerations are included in Section 4.9 of the EIS and S.4.9 of the Summary.

Basis for the Preferred Alternative - The elements of the preferred alternative discussed above have been selected based on the following considerations:

1. ***Management Alternative*** - The various management alternatives considered are discussed in Sections 2.2 through 2.4 of the EIS and S.2.2, S.2.4 and S.2.5 of the Summary. The analyses in Sections 4.2 through 4.5 of the EIS and S.4.2, S.4.3, S.4.5 and S.4.6 of the Summary demonstrate that the impacts on the environment, involved workers, or the citizens of the United States from implementation of any of the management alternatives or implementation alternatives analyzed (other than beneficial impacts associated with support for United States nuclear weapons nonproliferation policy) would be small and completely within the applicable regulatory limits, and would not provide a basis for discrimination among the alternatives. As a result, the process for selection of the elements of the preferred alternative focused on programmatic considerations:

    a. DOE and the Department of State concluded that the No Action Alternative and Management Alternative 2, Implementation Alternative 1a (Overseas Storage) would be unacceptable since these alternatives are not consistent with United States nuclear weapons nonproliferation policy objectives.

    b. DOE and the Department of State believe that the basic implementation of Management Alternative 1 would be undesirable to the extent that it would involve indefinite storage of foreign research reactor spent nuclear fuel in a form that is not suitable for disposal. Management Alternative 1 modified to rely solely on Implementation Alternative 6 (Near Term Conventional Chemical Separation in the United States) would raise nuclear weapons nonproliferation policy questions. Management Alternative 1 modified to rely solely on Implementation Alternative 7 (Developmental Treatment and/or Packaging Technologies) could not be selected at this time because no decision has been made on which technology will be pursued.

    c. DOE and the Department of State also believe that Management Alternative 2, Implementation Alternative 1b (Overseas Reprocessing) would be technically complex and potentially extremely expensive because it would require the United States to accept reprocessing wastes from the overseas reprocessing operations. This is due to the fact that both of the countries in which the overseas reprocessing might be accomplished require the reprocessing wastes to leave their countries, and many of the countries that would be covered by the proposed policy cannot accept the return of such reprocessing wastes. The intermediate-level radioactive wastes produced in Europe during reprocessing of research reactor spent nuclear fuel are often in a concreted waste form, unlike any high-level radioactive waste form in the United States. This concreted waste form has not been evaluated for disposal in a United States geologic repository. Accordingly, acceptance of such waste in the United

States likely could require expensive, currently unproven treatment and/or packaging technologies to transform it into a form that would be acceptable for disposal.

d. The sample hybrid alternative (Management Alternative 3) analyzed in the Draft EIS involved partial reprocessing overseas coupled with partial management in the United States. In order for this alternative to be consistent with United States nuclear weapons nonproliferation policy objectives, certain conditions would have to be met by either the reprocessor (e.g., Dounreay) or the research reactor operators. Staff from both DOE and the Department of State have addressed this issue with representatives of the United Kingdom Department of Trade and Industry and reactor operators, and have determined that it would not be possible to ensure compliance with the United States nuclear weapons nonproliferation policy objectives. The primary concern was the inability to ensure that any separated HEU would be blended down to LEU. Obtaining the reactor operators' agreement to such a policy would likely require significant financial subsidies. The potential cost of achieving agreement to blend down the uranium, plus uncertainties regarding Dounreay's long-term availability, led DOE and the Department of State to conclude that successful implementation of this alternative could not be relied on.

None of the alternatives analyzed in the Draft EIS could be implemented without some degree of difficulty. However, a modification of Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States), incorporating a combination of alternatives to the basic implementation components balances policy, technical, cost and schedule requirements. DOE and the Department of State consider that this approach provides the highest assurance that programmatic requirements could be met. This combination also provides the strongest support for United States nuclear weapons nonproliferation policy objectives as all aspects of the alternative would be under the control of DOE, either directly or through the spent nuclear fuel acceptance contracts with the reactor operators.

2. ***Management Technology*** - The alternative spent nuclear fuel management technologies considered are discussed in Sections 2.2.2.7 and 2.6.5 of the EIS and S.2.2.1.9 and S.2.2.2 of the Summary. The approaches fall into four broad categories, as follows:

Wet Storage - Wet storage is a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if it were used to implement the proposed action. Furthermore, DOE currently has wet storage facilities in operation at the Savannah River Site and the Idaho National Engineering Laboratory that could be used for storage of foreign research reactor spent nuclear fuel. However, wet storage requires attention to ensure that the storage conditions do not foster slow degradation of the spent nuclear fuel through corrosion.

Dry Storage - Dry storage is also a proven technology, that would also have no more than small impacts, completely within the applicable regulatory limits, if used to implement the proposed action. It is the storage medium that is being selected at all commercial power reactor sites where additional storage capacity is being built. However, it has not been used for research reactor spent nuclear fuel in the United States. Dry storage capacity could be provided at the management sites in time to meet the program's projected needs, if initial spent nuclear fuel receipts were placed into the available wet storage.

Chemical Separation - Chemical separation is also a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if used to implement the proposed action. However, DOE is phasing out its chemical separation activities and is currently reprocessing only at the Savannah River Site to stabilize materials for health and safety reasons. Because these chemical separations facilities could be used to treat the foreign research reactor spent nuclear fuel, they provide a contingency to be considered pending availability of an alternate means of treating and/or packaging the spent nuclear fuel prior to ultimate disposition.

New Technologies - Due to concerns regarding geologic disposal of intact spent fuel containing HEU (i.e., the possibility of uncontrolled criticality incidents), some form of treatment of this spent nuclear fuel may be required. While several concepts have been proposed for new treatment and/or packaging technologies, none of them are ready for implementation at this time. Prior to a decision leading to their implementation, additional development work would be required to determine whether and how they could be implemented, based on technical and cost considerations.

In order to effectively implement the preferred alternative of accepting and managing the foreign research reactor spent nuclear fuel in the United States, DOE and the Department of State developed the three point strategy for management of aluminum-based spent nuclear fuel discussed earlier in this Section. This strategy draws on the strengths of each of the spent nuclear fuel management technologies discussed above, while avoiding sole reliance on any of them. Due to the relatively more robust nature of the TRIGA spent nuclear fuel, DOE believes that minimal additional development may be needed to prepare it for storage and final disposition. Accordingly, the preferred alternative specifies that the TRIGA spent nuclear fuel would be placed in existing dry storage facilities at the Idaho National Engineering Laboratory. However, the program to qualify the final geologic disposal form for the TRIGA spent nuclear fuel will continue and the appropriate treatment, if any, would be identified and implemented.

3. ***Policy Duration*** - The alternative policy durations considered are defined in Sections 2.2.2.1 and 2.2.2.2 of the EIS and S.2.2.2 of the Summary. Analysis of these alternatives concluded that the 5-year option is likely to provide insufficient time for the reactor operators to arrange for alternative spent nuclear fuel disposal mechanisms, and thus might result in some reactor operators refusing to cooperate fully with United States nuclear weapons nonproliferation programs. This, in turn, could undermine international cooperation with other nuclear weapons nonproliferation programs the United States might seek to implement.

On the other hand, the analysis determined that there was insufficient benefit to be gained from indefinite acceptance of all the spent nuclear fuel containing HEU because such an approach likely would provide insufficient incentive for other countries to proceed expeditiously with arrangements for alternative disposal mechanisms not involving the United States.

The approach incorporated into the preferred alternative allows sufficient incentive to the reactor operators to ensure their cooperation, while specifying a definite cut-off point. This alternative provides sufficient lead time to allow the reactor operators to make other arrangements for disposition of their spent nuclear fuel, and provides sufficient time to accept all spent nuclear fuel containing HEU enriched in the United States.

4. ***Amount of Material to Manage*** - The alternative amounts of material that might be covered by the proposed policy are defined in Sections 2.2.1.3 and 2.2.2.1 of the EIS and S.2.2.2 of the Summary. DOE and the Department of State concluded that management of spent nuclear fuel only from other-than-high-income economy countries would strongly encourage the resurgence of the use of HEU in the high-income economy countries, as well as opening the United States, fairly or unfairly, to charges that we are not living up to our commitments under the *Treaty on the Non-Proliferation of Nuclear Weapons*. Management of only spent nuclear fuel containing HEU would penalize those reactors that have already converted to the use of LEU fuel, and would provide an incentive for reactors to continue to use HEU fuel, or switch back to its use. The impacts that would result from management of any of these different amounts of material would be small, and within the applicable regulatory limits.

   DOE and the Department of State concluded that management of all of the aluminum-based and TRIGA foreign research reactor spent nuclear fuel currently in storage or projected to be discharged during the policy period, and target material containing uranium enriched in the United States, would provide the best support for the objectives of the proposed policy. Implementation of this preferred alternative would provide an opportunity for removal of the maximum amount of HEU from civil commerce and would provide an incentive for the continued conversion to and use of LEU as fuel for foreign research reactors, in place of highly-enriched (weapons-grade) uranium.

5. ***Marine Transport*** - The alternative approaches to marine transport of foreign research reactor spent nuclear fuel are discussed in Section 2.2.1.5 of the EIS. The analysis in the EIS demonstrates that the impacts to the environment, workers, or the public from transport of the spent nuclear fuel using any of these types of ships would be small, and within the regulatory limits. The analyses do not identify any difference in the small impacts that would result from the use of purpose-built vs. general purpose ships. Since "military transports" are in fact the same type of ship as the chartered commercial cargo ships and are crewed by civilians, use of "military transports" would not actually result in any difference in impacts. DOE and the Department of State believe that use of actual warships would be both unnecessary from a security standpoint and could entail additional risk to the environment and the public, since such ships do not routinely carry cargo.

The approach selected by DOE and the Department of State for the preferred alternative provides maximum flexibility for marine transport.

6. ***Ground Transport*** - The ground transportation alternatives are defined in Section 2.2.1.7 of the EIS and S.2.2.1.7 of the Summary. The analyses in the EIS demonstrate that the impacts to the environment, workers, or the public, from any of these modes of ground transport (counting barge as a mode of "ground transport") would be small and within the applicable regulatory limits. Furthermore, the differences in potential impacts between the truck, rail and barge alternatives were not significant.

   Both the truck and rail transportation options have been used successfully to transport foreign research reactor spent nuclear fuel in the past. Truck transport was the predominant mode used for over twenty years, until the old "Off-Site Fuels Policy" lapsed in 1988. Rail was the mode used for both shipments under the *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*. Since neither of the preferred ports of entry (see item 8 below) can reasonably provide barge transport to either of the proposed management sites, barge transport was dismissed from consideration in the preferred alternative.

   By providing for either truck or rail transport, the preferred alternative would build on previous satisfactory experience while providing maximum flexibility for dealing with changes in the transportation process in the future.

7. ***Title Transfer Location*** - The alternative points at which DOE might take title to the spent nuclear fuel and target material are discussed in Sections 2.2.1.4 and 2.2.2.4 of the EIS and S.2.2.2 of the Summary. The point at which title would be transferred has no effect on the physical processes that would take place, and thus would not have any effect on the impacts on the environment, workers, or on the public. The Price-Anderson Act would provide liability protection in the unlikely event of a nuclear accident in the United States, whether or not DOE had taken title to the spent nuclear fuel at the time of such an accident. As a result, DOE and the Department of State concluded that the selection of the title transfer location could be made solely on programmatic considerations.

   Acceptance of title at the foreign research reactor sites could make the United States Government liable for any accident that might occur in the country of origin, or on the high seas. DOE and the Department of State have been unable to identify any advantage to the United States of taking title outside the United States.

   Taking title at the limit of United States territorial waters makes the title transfer depend solely on when the ship enters United States waters, which could be difficult for DOE to control in certain circumstances (e.g., a storm).

   Acceptance of title when the foreign research reactor spent nuclear fuel actually enters the land mass of the United States provides the most certainty for implementation.

The approach incorporated into the preferred alternative ensures that liability for accidents during the transportation process outside the United States would remain with the reactor operators while reinforcing in the minds of the public that the United States Government would be accountable in the unlikely event of an accident within United States territory.

8. ***Ports of Entry*** - The alternative ports of entry considered are discussed in Sections 2.2.1.6 and 3.2 of the EIS and S.2.2.1.6 of the Summary. The analyses in the EIS demonstrate that the impacts on either the environment, workers, or the public due to use of any of the potential ports of entry analyzed would be small and within applicable regulatory limits.

   Although any one or all of the ten ports of entry described in Sections 2.2.1.6 and 3.2 of the EIS would be acceptable ports of entry, DOE and the Department of State concluded that foreign research reactor spent nuclear fuel marine shipments to the United States should be made via the military ports (selected from among those analyzed in the EIS and found acceptable) in close proximity to the spent nuclear fuel management sites. DOE would seek to transport multiple casks per ship to keep the total number of shipments as low as possible, as well as to reduce risks. The exact number of shipments that might be made would be determined by several factors that are unknown at this time, such as the times at which the reactor operators need to make shipments over the 13 year shipping period, the geographic distribution of the reactors, and the availability of suitable ships that would stop at the required ports to pick up and drop off the spent nuclear fuel and target material.

   Use of military ports would provide additional confidence in the safety of the shipments due to the increased security associated with the military ports. It could also require much of the spent nuclear fuel to be shipped via chartered ships since commercial ships would not have stops scheduled at military ports, increasing the cost of spent nuclear fuel shipping. This additional cost would be borne by the reactor operators for shipments from high-income economy countries, and by the United States for shipments from other-than-high-income economy countries. Additional costs would be kept to a minimum by shipping as many casks as possible on each ship (up to a maximum of eight per ship).

9. ***Management Sites*** - The question of which sites should be used for management of all of DOE's spent nuclear fuel was addressed in the Programmatic SNF&INEL Final EIS (DOE, 1995c). That EIS included consideration of the potential receipt of the foreign research reactor spent nuclear fuel. The Record of Decision for that EIS, issued on May 30, 1995, specifies that any aluminum-based foreign research reactor spent nuclear fuel accepted in the United States shall be managed at the Savannah River Site; and that the remaining foreign research reactor spent nuclear fuel shall be managed at the Idaho National Engineering Laboratory. The site for management of the target material was left to be decided under this EIS. All of the target material currently in DOE's possession is managed at the Savannah River Site. The approach incorporated into the preferred alternative is in compliance with the decision specified in the Record of Decision for the Programmatic SNF&INEL Final EIS.

The analyses in the EIS demonstrate that the impacts to either the environment or the public through use of any of the sites for management of the foreign research reactor spent nuclear fuel and target material would be small, and within the applicable regulatory limits.

10. ***Financing Arrangement*** - The alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS and S.2.2.2 of the Summary. The financing arrangement used for the proposed action would have no effect on the physical processes that would take place, and thus would not have any effect on the potential impacts on the environment, or on the public. However, it could affect how many foreign research reactor operators elect to ship spent nuclear fuel to the United States. For instance, if DOE and the Department of State chose to charge a full cost recovery fee to all reactors, many, if not all, of the reactors in other-than-high-income economy countries would not have the financial resources to participate. On the other hand, if the United States subsidized all of the reactors, the United States would bear the full financial burden, even for reactors which can afford to pay their fair share.

    DOE and the Department of State concluded that, to ensure that reactor operators in other-than-high-income economy countries would participate in the program, the United States should subsidize receipt of their spent nuclear fuel. DOE and the Department of State also concluded that DOE should strive to recover as much of the cost of managing the spent nuclear fuel as possible from high-income economy countries. DOE concluded that it would announce the fee in a *Federal Register* notice, so that the fee may be changed from time to time as necessary to reflect inflation or improvements in DOE's knowledge concerning the costs of the activities to be carried out.

    Such an approach would encourage participation by as many other-than-high-income economy countries as possible, would recover as much as possible of the United States' expenses for management of spent nuclear fuel from high-income economy countries without encouraging any of them to resort to reprocessing of their spent nuclear fuel, and would provide a mechanism through which to account for inflation and future definition of program details.

S.2.4 Management Alternative 2 - Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Under this Management Alternative, DOE and the Department of State would seek to facilitate the management of foreign research reactor spent nuclear fuel overseas in a manner that would be consistent with U.S. nuclear weapons nonproliferation policy. DOE and the Department of State have evaluated two subalternatives: Overseas Storage and Overseas Reprocessing.

1a. Overseas Storage

The United States would assist foreign research reactors that are able to store their spent nuclear fuel in facilities in their own countries as a step toward final disposition. U.S. assistance would be provided to ensure that appropriate storage technologies, regulations and safeguards were applied.

1b. Overseas Reprocessing

The United States would facilitate and provide nontechnical (financial and/or logistical) assistance to foreign research reactors and reprocessors to facilitate reprocessing of spent nuclear fuel overseas in facilities operated under international safeguards consistent with U.S. nuclear weapons nonproliferation concerns.

The overseas reprocessing option was evaluated in light of the U.S. nuclear weapons nonproliferation policy on HEU minimization. For example, factors such as the following were considered:

- A commitment that HEU separated during reprocessing would be blended down to LEU for research reactors which are converting to LEU.
- The foreign reprocessors would provide the capability to reprocess LEU as well as HEU.
- Research reactors would be encouraged to convert to LEU if a LEU fuel exists or is developed that will allow such operation.

Arrangements would have to be worked out with foreign reprocessors that would be consistent with U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

S.2.5 Management Alternative 3 - Combination of Elements From Management Alternatives 1 and 2 (Hybrid Alternative)

In implementing the proposed action, DOE and the Department of State could combine implementation elements from Management Alternatives 1 and 2, such as partial storage or reprocessing overseas with partial storage or chemical separation in the United States.

To demonstrate the kind of hybrid alternatives that could be developed, this EIS considers the following hybrid alternative example: DOE and the Department of State would facilitate the reprocessing of foreign research reactor spent nuclear fuel at Western European reprocessing facilities (e.g., Dounreay or Marcoule) for research reactors in countries that could accept the waste from reprocessing, and DOE would accept and manage in the United States the rest of the foreign research reactor spent nuclear fuel from countries that could not accept the waste from reprocessing. Of the foreign research reactor spent nuclear fuel to be accepted in the United States, the aluminum-based portion would be chemically separated at the Savannah River Site and the TRIGA portion would be stored in existing facilities at the Idaho National Engineering Laboratory.

The impacts to the U.S. environment from hybrid alternatives would be covered by the analyses presented in the EIS for Management Alternative 1, because the analyses for Management Alternative 1 consider the maximum amount of foreign research reactor spent nuclear fuel that could be accepted, stored, and/or chemically separated in the United States.

S.2.6 No Action

In the No Action Alternative, the United States would neither manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States, nor provide technical assistance or financial incentives for overseas storage or reprocessing. In this case, there would be no foreign research reactor spent nuclear fuel shipments to the United States and no assistance to foreign countries for managing foreign research reactor spent nuclear fuel overseas.

S.2.7 Characteristics of Spent Nuclear Fuel Management

This section briefly summarizes information on the characteristics of the spent nuclear fuel to be managed, the types of transportation casks considered, management site storage facilities, chemical separation facilities in the United States, and foreign reprocessing facilities.

S.2.7.1 Characteristics of Foreign Research Reactor Spent Nuclear Fuel

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. Spent nuclear fuel is radioactive because of the presence of the radioactive isotopes, products of the fission process. The radiation of most concern from spent nuclear fuel is gamma rays. Although the radiation levels can be very high, the gamma ray intensities are readily reduced by shielding the fuel elements with such materials as steel, lead, concrete, and water during the various phases of handling, transporting, or storing the spent nuclear fuel elements.

An issue associated with the management of spent nuclear fuel containing significant amounts of fissionable material is the potential for a self-sustaining nuclear fission process called criticality. Prevention of criticality conditions enters in the design of the spent nuclear fuel transportation casks, the spent nuclear fuel storage and processing facilities, and the spent nuclear fuel packaging for ultimate disposition. In general, criticality prevention is accomplished by either controlling the amount of fissionable material present within a certain volume (dilution or spatial separation techniques) or by introducing neutron absorbing materials that reduce the number of neutrons available to the fission process (poisoning technique).

Two types of foreign research reactor spent nuclear fuel are covered under the proposed policy. They are the aluminum-based fuel and TRIGA-type reactor fuel. In addition to the two types of spent nuclear fuel described above, target material is also covered under the proposed policy. Target material is the residual material from medical isotope production targets irradiated in research reactors.

Figure S-8 graphically depicts the differences in size of a typical pressurized water reactor assembly, a typical aluminum-based fuel element, and a TRIGA fuel element.

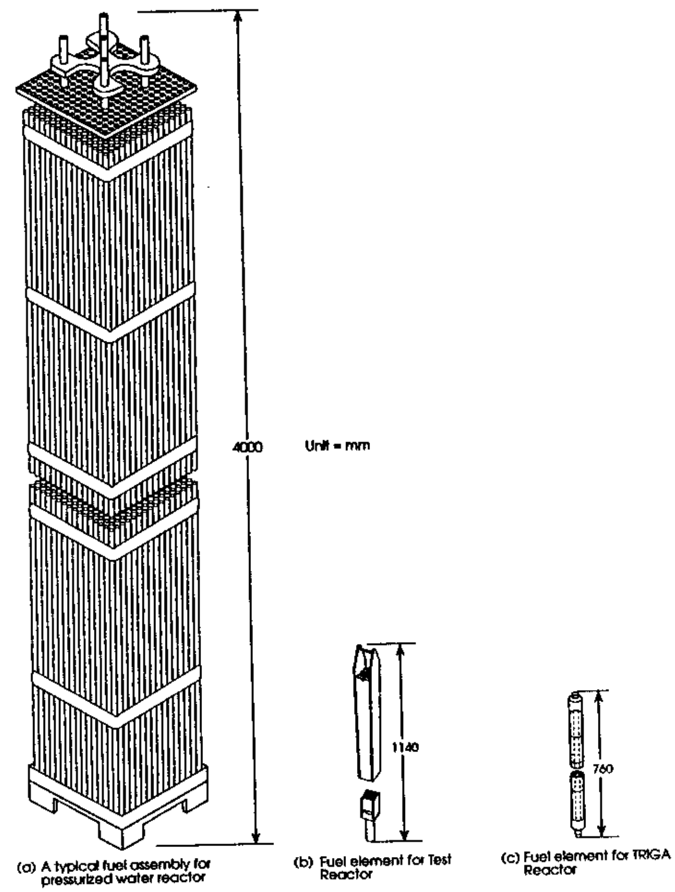


Figure S-8 Typical Spent Nuclear Fuel Elements

S.2.7.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages, usually weighing several tons, called transportation casks. A typical cask for the transportation of foreign research reactor spent nuclear fuel elements is shown in Figure S-9.

The casks are designed to provide shielding from radiation. However, a low radiation field is present outside the cask — frequently less than one millirem (mrem) per hour at one meter (3.3 ft) away from the cask. A full cask can carry from 13 to 120 spent nuclear fuel elements from foreign research reactors, depending on fuel element design, size, and cask capacity. The casks that would be used to transport foreign research reactor spent nuclear fuel to the United States are "Type B" casks designed on the basis of international regulations essentially identical to those promulgated by the NRC and certified by the Department of Transportation. "Type B" casks have been used for years to transport spent nuclear fuel elements within the United States and around the world. In more than

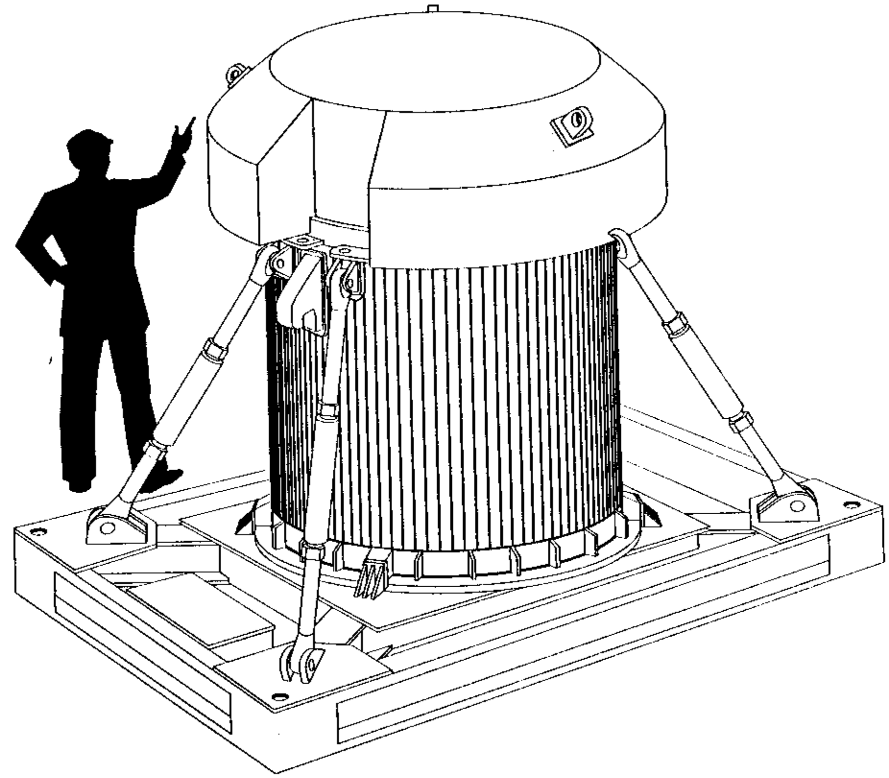


Figure S-9 Typical Spent Nuclear Fuel Transportation Cask

four decades of transporting spent nuclear fuel within the United States, no accident has ever occurred in which a "Type B" spent nuclear fuel transportation cask was punctured or spent nuclear fuel contents released, even in actual highway accidents.

S.2.7.3 Spent Nuclear Fuel Storage Facilities in the United States

The EIS analyzes a variety of scenarios in which each site could manage foreign research reactor spent nuclear fuel. However, as noted in S.2.2.1.8, in accordance with the Record of Decision for the Programmatic SNF&INEL Final EIS, all of the aluminum clad foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory, pending ultimate disposition. Of the five management sites considered in the Draft EIS, only the Savannah River Site and the Idaho National Engineering Laboratory have facilities that could be available in 1996. The other three could become available as management sites at a later date after construction or refurbishment of appropriate facilities could be completed. This constraint has resulted in a two-phased approach to the implementation of the policy. For the purpose of site impact analysis, the implementation of the policy was divided into two functional periods -- the period during which receipt and management of foreign research reactor spent nuclear fuel would be accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities could be used (Phase 2). The following discussion summarizes key points concerning facility capabilities and assumptions at each site, which drive the analysis of environmental impacts in the EIS.

S.2.7.3.1 Savannah River Site

As a potential Phase 1 storage site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at its existing wet storage facilities. The Receiving Basin for Offsite Fuels and the L-Reactor Disassembly Basin are considered for this purpose.

As a potential Phase 2 storage site, the Savannah River Site could continue to receive foreign research reactor spent nuclear fuel in a new dry storage facility or a new wet storage facility that would be constructed in the H-Area of the site or a refurbished Barnwell Nuclear Fuels Plant which would have to be acquired by DOE. The spent nuclear fuel would be managed at the new storage facility until ultimate disposition.

S.2.7.3.2 Idaho National Engineering Laboratory

As a potential Phase 1 storage site under Management Alternative 1, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and/or wet storage facilities. The existing facilities identified for this purpose would be the Fluorinel Dissolution and Fuel Storage Facility in CPP-666, the Irradiated Fuel Storage Facility in CPP-603, and the CPP-749 storage area.

As a potential Phase 2 storage site, the Idaho National Engineering Laboratory could continue to receive and manage foreign research reactor spent nuclear fuel at a new dry storage or wet storage facility to be constructed at the site.

S.2.7.3.3 The Hanford Site, Oak Ridge Reservation, and Nevada Test Site

The Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site could only be Phase 2 storage sites (under Management Alternative 1) if they had been selected as management sites under the Programmatic SNF&INEL Final EIS Record of Decision. As noted in Summary Section S.1.4, these three sites are no longer candidates for management of the foreign research reactor spent nuclear fuel under the Record of Decision in the Programmatic SNF&INEL Final EIS, but are considered in this EIS in order to maintain consistency with the analyses provided in the Programmatic SNF&INEL Final EIS.

S.2.7.4 Chemical Separation Technology and Facilities in the United States

The EIS evaluates near-term conventional chemical separation in the United States as an alternative method of managing foreign research reactor spent nuclear fuel. Chemical separation involves separating the uranium in the spent nuclear fuel from the other material (i.e., cladding material, fission products, etc.). Aluminum would be the predominant cladding material. Waste materials would mainly be fission products, and consist of radioactive species such as cesium and strontium. The separated uranium could be placed into commerce as new fuel (as LEU fuel) or could require further disposition steps. Vitrification (conversion into a solid glass form) of the high-level waste would be the preferred waste management approach.

An aqueous chemical method is the only processing method applied on a large scale. All existing chemical separation plants use an extraction process that has been in use for some 40 years. Under the chemical separation implementation alternative of Management Alternative 1, foreign research reactor spent nuclear fuel would be chemically separated at the Savannah River Site or the Idaho National Engineering Laboratory. For purposes of analysis, this EIS assumes that the Savannah River Site would chemically separate aluminum-based spent nuclear fuel in the F-Canyon and the Idaho National Engineering Laboratory would chemically separate both aluminum-based and TRIGA spent nuclear fuel. Near-term conventional chemical separation of foreign research reactor spent nuclear fuel at the other three proposed foreign research reactor spent nuclear fuel management sites would not be considered since the Oak Ridge Reservation and the Nevada Test Site do not have facilities in which such chemical separation could be conducted, and the facilities at the Hanford Site are no longer operable. Figure S-10 provides an overview of chemical separation.

S.2.7.5 Foreign Reprocessing Facilities

Both France and the United Kingdom have modern fuel cycle facilities and offer reprocessing services to international customers. These facilities are capable of reprocessing spent nuclear fuel and preparing the waste products for disposal. Both France and the United Kingdom would require the country operating the reactor to accept the waste from reprocessing.

S.2.8 Emergency Management and Response

An emergency management and response infrastructure exists to support the implementation of those Management Alternatives that would be carried out in the United States, including ports of entry, ground transport routes, and management sites. In the United States, State and local governments are required to have emergency management and response programs. These programs must be capable of managing all hazards, ranging from natural disasters to hazardous material incidents on a day-to-day basis. These programs include support from special emergency response teams and emergency operations centers.

S.2.9 Security Measures

Domestic transportation of foreign research reactor spent nuclear fuel would be under the regulatory jurisdiction of the Department of Transportation and the NRC. In the event that foreign research reactor spent nuclear fuel was transported through a military port of entry, applicable requirements would be established in advance by the U.S. Department of Defense, DOE, and NRC to provide the appropriate level of security.

The objectives of the security measures during transportation of spent nuclear fuel are to minimize the possibilities for sabotage of spent nuclear fuel shipments, and facilitate the location and recovery of spent nuclear fuel shipments in the unlikely event that a shipment came under the control of unauthorized persons. Specific elements of the security

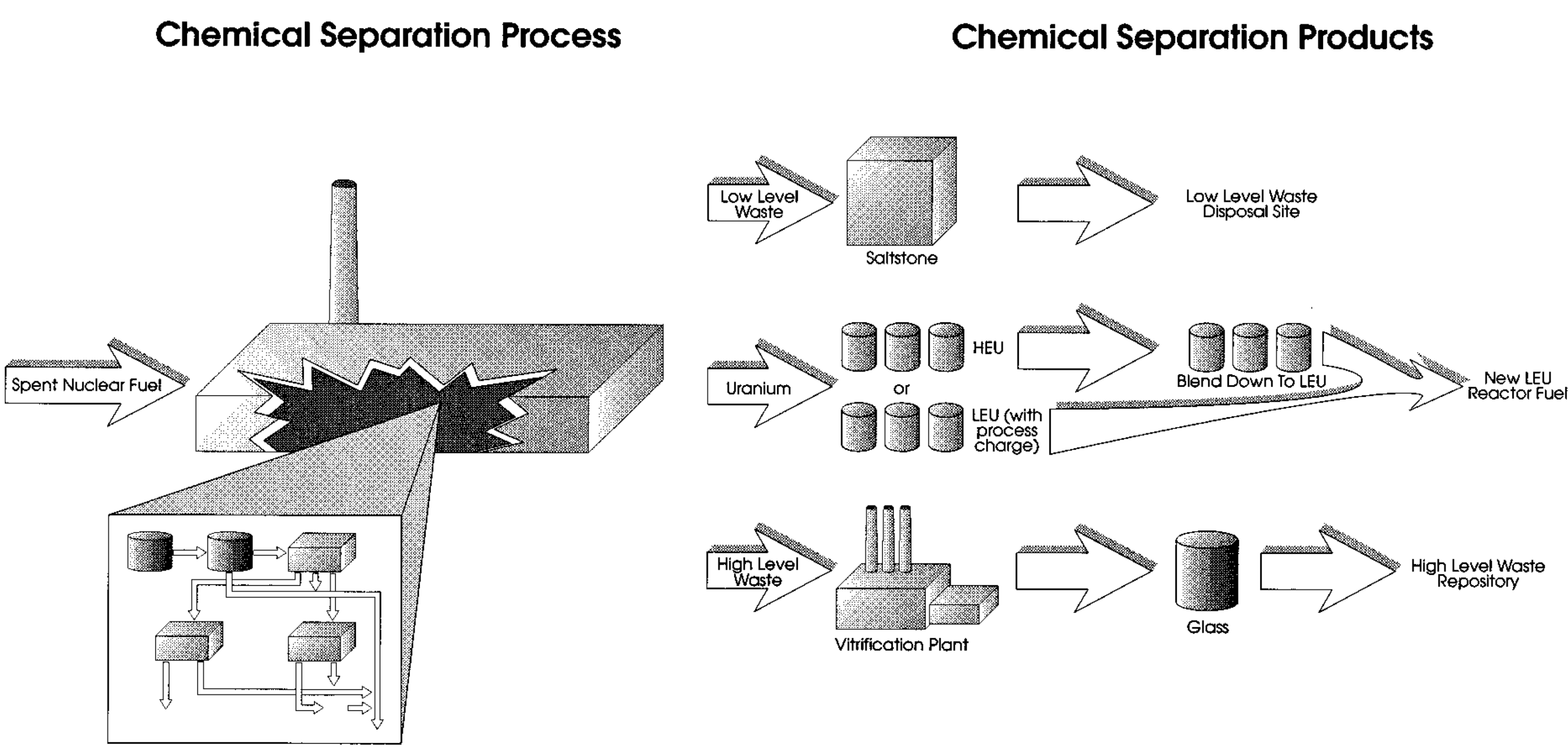


Figure S-10 Chemical Separation Overview measures to be implemented would be included in the Transportation Plan developed by DOE in consultation with State, local, and Tribal officials prior to any actual spent nuclear fuel shipments.

S.2.10 Additional Alternatives Considered but Dismissed from Detailed Analysis

The EIS considered additional alternatives that were dismissed as unreasonable and therefore were not further analyzed. These are the use of an air mode of transportation and acceptance of foreign research reactor spent nuclear fuel only from countries that present an actual nuclear weapons nonproliferation risk.

The air mode of transportation was not considered to be a feasible alternative to the sea mode for transportation of the foreign research reactor spent nuclear fuel for two reasons. First, there is no commercial operational experience in the United States with air transport of spent nuclear fuel. Second, no spent nuclear fuel transportation cask has been certified to meet air transport packaging standards.

Accepting foreign research reactor spent nuclear fuel only from countries posing an actual nuclear weapons nonproliferation risk would not fully address the key U.S. nuclear weapons nonproliferation goal of the proposed policy--namely, to reduce and eventually eliminate the use of HEU in research reactors worldwide.

S.3 Affected Environment

The proposed action would potentially affect marine, port, transportation route, and management site environments. Chapter 3 and Appendices A and E of the EIS describe these potentially affected environments. Geological, chemical, physical, and biological descriptions of the oceans are included in Chapter 3 to provide a background for the evaluation of marine environmental effects that would result from implementation of the proposed policy. Demographic data and description of the natural environment surrounding candidate ports of entry and management sites follow the description of the marine environment. The EIS also provides a description of populations residing near representative ground and water transportation routes which could be used to transport foreign research reactor spent nuclear fuel from candidate ports of entry to management sites.

S.4 Policy Considerations and Environmental Impacts

The EIS assesses the policy considerations and potential environmental impacts resulting from each of the Management Alternatives for implementation of the proposed policy, including the preferred alternative designated by DOE and the Department of State, and the No Action Alternative. Policy considerations are addressed by characterizing the extent to which each of the alternatives supports the U.S. goal of nuclear weapons nonproliferation. This characterization takes two forms: 1) estimates of the maximum amount of HEU that could be removed from international commerce under each alternative, and 2) the extent to which each alternative would provide incentives for foreign research reactor operators to convert their reactors to LEU fuel.

The environmental analyses address potential impacts to workers, the public, and the environment. The analyses are based on conservative assumptions (that is, those that tend to overstate the risks). In other words, the analytical approaches are designed to produce estimates of the maximum reasonably foreseeable risks. Cumulative impacts were determined by evaluating past, present, and reasonably foreseeable DOE and non-DOE related activities, in combination with the alternatives. Radiological impacts were calculated in terms of absorbed dose and associated health effects in the exposed populations. Nonradiological impacts to the environment, namely land use, waste management, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, noise, utilities and energy, and socioeconomics were also analyzed in this EIS. This analysis reveals that none of the impacts clearly differentiates among the potential management sites and the environmental impacts are estimated to be low. Environmental justice concerns are addressed in the EIS by characterizing the distribution of minority and low-income households near candidate ports, along transportation routes, and near management sites. Based on the analyses in the EIS, the health and environmental effects for the total population, including low-income and minority populations, were found to be very low.

Implementation of the proposed action would have little effect on the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports, along transportation routes, and residing near management sites. The EIS analyses show that economic benefits from increased cargo handling, transportation, and storage at management sites would be small for the general population or any particular segment of the population residing near ports, transportation routes and management sites.

S.4.1 Overview of Environmental Impacts

The EIS presents impacts from potential radiological and nonradiological activities and accidents in four segments of the potentially affected environment. These four segments are:

- Marine transport
- Port activity
- Ground transport, and
- Management site activity.

The conclusion in each of the segments individually, and in the four segments collectively, is that the implementation of the proposed action would present low risks to workers and to the public. Marine transport, port, ground transport, and management site impacts are addressed in more detail in the discussion of impacts for each of the Management Alternatives.

S.4.1.1 General Radiological Health Effects

One way of presenting potential impacts to human populations in the EIS is by using radiation dose. Potential damage to human cells from radiation is measured in rem and millirem (mrem). The U.S. government has set a limit of 5,000 mrem (5 rem) per year for individual radiation workers and 100 mrem (0.1 rem) per year for individual members of the public from man-made, non-medical sources. The average American receives about 300 mrem of radiation per year from natural sources such as radon gas from the earth's soil. Living in a brick house rather than a wood-frame house can add 45-50 mrem annually to one's dose. Living at high altitudes rather than at sea level also increases one's dose. A single coast-to-coast flight exposes an individual to about 4 mrem.

Another way of presenting results in the EIS is by using the concept of risk. The most significant radiation-related illness is the inducement and development of cancers that may lead to death in later years. This effect is called a latent cancer fatality. The risks of incurring a latent cancer fatality are estimated by converting radiation doses into possible numbers of future cancer fatalities.

For an exposed population group, the latent cancer fatality number is the chance that there would be an additional latent cancer fatality within the exposed group. The chance that a member of that group would develop a latent cancer fatality depends on the size of the exposed group. For example, if the estimated number of latent cancer fatalities for a group of 100,000 people is one, the average member of this group would have a one in 100,000 chance of developing a latent cancer fatality.

Radiological risk can also be expressed for hypothetical individuals who could record the highest possible dose in a given situation. Examples are a seaman who inspects the casks at sea, a port worker who unloads the casks, a truck driver who transports the casks to a management site, or an individual living at the site boundary of a management site. When a latent cancer fatality number is given for an individual, it represents the chance that the exposed individual would develop a latent cancer fatality. As a practical matter, the maximally exposed individual during incident-free operations

Measuring Radiation Exposure

Potential radiological impacts are estimated for the highest radiation dose any single person might receive, as well as the collective dose a particular population might receive, such as all those living in the vicinity of a port. Two primary units of radiation dose measurement are used in the Final EIS to estimate these impacts: the rem and person-rem.

The rem is a unit of radiation dose. Because 1 rem is a relatively large dose, the unit actually used most frequently is the millirem (mrem), which is equal to 1/1000 of a rem. It is estimated that the average individual in the United States receives a background dose of about 300 mrem/yr from all natural sources including radon.

Radiation dose to a population or a group of persons is measured in person-rem. The total population dose (all the person-rem) is determined by adding all the individual doses in the exposed group. This measurement is particularly important when trying to take into account the potential impacts of very small doses on very large populations (for example, all those living along a transportation route).

Using a conversion factor, the estimated doses can be converted into possible numbers of health effects. Because the doses predicted in this study are far less than those known to cause immediate illness or fatality, only delayed health effects would occur. A delayed effect is measured in latent (future) cancer fatalities. For the general population, a collective dose of 2,000 person-rem is estimated to result in one additional latent cancer fatality within the affected population group.

Latent Cancer Fatalities Caused by Natural Background Radiation for an Individual Member of the General Public

Dose: Radioactivity from all natural sources combined produces about a 300 mrem (0.3 rem) dose to the average individual per year.

Probability: The probability of continuous exposure to this average dose is one.

Average Life Span: 72 years is considered to be the average lifetime.

Latent Cancer Fatalities Caused per Rem for an Individual Member of the General Public: 0.0005 latent cancer fatalities are estimated to be caused by exposure to 1 rem.

Calculation: Dose rate x life span x cancers caused per rem = 0.3 rem/yr x 72 yr x 0.0005 latent cancer fatalities per rem = 0.01 latent cancer fatalities per individual lifetime.

Risk: Probability x latent cancer fatalities = 1 x 0.01 = 0.01 latent cancer fatalities, which is about 1 chance in 100 of death from exposure to natural background radiation over a lifetime.

would be a worker because he or she would be close to the spent nuclear fuel. If necessary, DOE would implement mitigation measures to maintain individual doses under the regulatory limit for the general public. The doses and risks estimated in the EIS reflect DOE mitigation efforts directed at ship crews, port workers, and truck drivers.

Radiological risks calculated in the EIS are also compared to those of common activities, such as smoking, flying, or receiving a medical x-ray.

S.4.2 Policy Considerations and Environmental Impacts of the Basic Implementation of Management Alternative 1

Under the basic implementation of Management Alternative 1, all the foreign research reactor spent nuclear fuel could be accepted into the United States. Up to 4.6 metric tons (5.1 tons) of HEU would be removed from international commerce. DOE and the Department of State believe implementation of this alternative would promote the nuclear weapons nonproliferation objective of reducing, and eventually eliminating, the use of highly-enriched (weapons-grade) uranium in civil programs worldwide. The spent nuclear fuel could be managed safely and securely at any of five management sites.

The following sections summarize the environmental impacts of the four segments of the affected environment under the basic implementation of Management Alternative 1.

S.4.2.1 Marine Transport Impacts

The shipment of foreign research reactor spent nuclear fuel would begin with the transport of the spent nuclear fuel from the onsite storage facility at the foreign research reactor to the foreign port. The spent nuclear fuel would then be shipped in transportation casks by sea (except for shipments from Canada) to a U.S. port. The potential impact of marine transport in the territorial waters of the United States was evaluated. Because implementation of the proposed action could involve ocean transport, the EIS also considers the environmental impacts on the global commons in accordance with Executive Order 12114. Shipments of any material via ocean transport entails risks to the ship's crew members and the environment. The risks result from transportation-related accidents and, in the case of radioactive materials, from exposure to the material itself.

S.4.2.1.1 Impacts of Incident-Free Marine Transport

The primary impact of incident-free marine shipping of foreign research reactor spent nuclear fuel would be upon the crews of the ships used to carry the spent nuclear fuel casks. Members of the general public and marine life would not receive any measurable dose from the spent nuclear fuel during marine transport. The crew would normally be separated from the cargo and shielded from radiation emitted from the cask by both the ship's structure and other cargo, resulting in small risk to the crew during most crew activities. Crew exposure would primarily be limited to crew members exposed during the loading and off-loading of the spent nuclear fuel casks and to crew members who would inspect the cargo daily to ensure secure stowage and operational safety of the vessel. This exposure from loading, inspection, and unloading of the casks would pose the highest radiation risk during incident-free marine transport.

An estimate of the maximum radiation dose that a member of a ship's crew might receive during an incident-free voyage of 21 days carrying foreign research reactor spent nuclear fuel is approximately 66 mrem. If this same crew member were to be involved in multiple voyages per year, then the yearly dose to this individual could exceed the DOE and NRC annual limit of 100 mrem per year for the public. Although this situation is not likely to occur, DOE would implement a system to track, through the contracted shippers, each ship and crew member involved in the shipment of foreign research reactor spent nuclear fuel. A clause in the contract for shipment of foreign research reactor spent nuclear fuel would require that any crew member approaching the 100 mrem per year limit be rotated to another job.

Nonradiological impacts were found to result in a small impact on the health of the public and workers. The number of shipments necessary to transport about 720 transportation casks would result in a minimal change in the number of ocean crossings by transport vessels. No increase in the exposure of the public to ship exhaust emissions or marine transport-related accidents is anticipated.

S.4.2.1.2 Impacts of Accidents During Marine Transport

The EIS analyzes two kinds of ocean accidents: 1) a ship collision, which in this EIS was assumed to result in damage to the cask and an on-board fire, and 2) loss of a cask at sea, where the cask sinks, and seawater penetrates the cask seals. However, the probability of a collision or fire resulting in a cask breach is low. The probability of a large radiation release is low because the spent nuclear fuel is a solid metal. In the type of collision or fire that could breach the cask and liberate significant quantities of radiation, the major impact on the crew would be the collision or fire, not the radiation. The radioactive particles dispersed over the ocean would not be in large enough amounts to have a measurable impact on the environment.

Immersing a cask in water does not cause the radioactive contents to be released immediately. Casks can be recovered in coastal waters and much deeper waters with modern technology. Thus, if a cask were to fall overboard in U.S. coastal waters or inland waters, DOE would employ modern underwater search techniques to locate and recover the cask, thus minimizing the potential impacts to marine life. Outside U.S. coastal waters, if a cask were to sink, modern technology would be used, if possible, in an effort to retrieve the cask. If the cask could not be recovered, seawater would penetrate the cask seals and corrode the spent nuclear fuel. There is no mechanism, however, by which the seawater entering the cask could be forced out of the cask. Thus, the radioactive material would escape from the cask at a very low rate and would have a very small effect on the marine environment.

S.4.2.2 Port Activities Impacts

Ports having high-, medium-, and low-population density and covering the Atlantic, Pacific, and Gulf coasts were analyzed. The risk of incurring latent cancer fatalities was found to be so low that the most likely outcome would be zero latent cancer fatalities due to accidents at ports. Calculations for incident-free and accident conditions clearly demonstrate that for the general population, including minority and low-income groups, the impacts would be very low. In consideration of environmental justice concerns, the EIS analyzed the characterization and distribution of minority and low-income households near candidate ports of entry. Minority and low-income populations living near the potential ports of entry would not be subjected to any greater impacts than the general population. Therefore, these populations would not receive disproportionately high and adverse impacts, but would be subject to the same very low impacts as would the general population.

Implementation of the proposed action would have few nonradiological effects on the environment at candidate ports, including the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports. The EIS analyses show that economic benefits resulting from increased cargo handling and transportation in the port area would be small for the general population, or any particular segment of the population, residing near candidate ports.

S.4.2.2.1 Impacts of Incident-Free Port Activities

The incident-free risks would predominantly be those to inspectors and port workers who would handle spent nuclear fuel casks. Based on the time to conduct port activities and the distances from the cask to the worker during these activities, a maximum dose (higher than the limit of 100 mrem per year) could result if the same individual inspected every shipment. This risk is not likely to occur, however, due to the fact that the same inspectors and port workers would not likely be responsible for all the shipments made in a given year. Nevertheless, DOE would mitigate this effect by implementing a system to track each inspector and port worker involved in the handling of foreign research reactor spent nuclear fuel to ensure that other inspectors or port workers would be used if any of these individuals approach a 100 mrem dose in any year.

S.4.2.2.2 Impacts of Accidents During Port Activities

Marine accidents could occur in the open ocean or in coastal passages. Taking into account the severity of the accident (i.e., severe collision with and without severe fires), the probability of the accident (i.e., the more severe the accident the less likely it is), the location of the accident (i.e., in the harbor channel or at the dock), meteorology, and nearby populations, the highest estimated risk of cancer for the entire population over the entire foreign research reactor spent nuclear fuel program is less than one in 10,000. This translates into less than one additional latent cancer fatality for the affected port population. The highest estimated risk to the maximally exposed individual of a future cancer death is less than one in a billion.

S.4.2.3 Ground and Barge Transport Impacts

Foreign research reactor spent nuclear fuel is transported in large, heavy transportation casks designed and constructed to contain radioactivity during severe transportation accidents. The NRC has estimated that transportation casks will withstand 99.4 percent of truck and rail accidents without breaching the cask. Only in severe accident conditions, which are of low probability, could the transportation cask be so damaged that there would be a reasonable possibility of release of radioactivity to the environment. Since 1949, there have been 21 incidents involving vehicles carrying irradiated fuel elements. None of these incidents resulted in damage to the structural integrity of the spent nuclear fuel transportation cask or release of the radioactive contents. The EIS calculations for incident-free and accident conditions demonstrate that for the general population the impacts would be low. Minority or low-income populations living near these routes would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts, but would be subject to low impacts as would the general population.

Impacts from barge transportation were also evaluated as a substitute for truck or rail transport. The only two locations where barge transport is feasible are from the Port of Portland, OR, up the Columbia River to the Hanford Site in Washington, and from the Port of Savannah, GA, up the Savannah River to the Savannah River Site in South Carolina. The net result is that the foreign research reactor spent nuclear fuel could be transported by barge with approximately the same level of risk to workers and the public as if it was transported by truck or rail. This level of risk is very low.

Implementation of the proposed action would have extremely low nonradiological effects on the environment along transportation routes, including the social and economic status of the general population, minority populations, and the low-income population residing along transportation routes. The EIS analyses show that economic benefits resulting from increased transportation of cargo along transportation routes would be small for the general population, or any particular segment of the population residing along transportation routes.

S.4.2.3.1 Impacts of Incident-Free Ground Transport

For incident-free ground transport, the radiological impacts result from the radiation field that surrounds the cask. Impacts are estimated for workers and the population along the transportation route. These impacts were quantified as the estimated number of radiation-related cancer fatalities and the estimated number of nonradiological fatalities from vehicular emissions and traffic accidents.

Allowing for transport by truck and/or rail, and assuming a wide range of inter-site shipments (depending on the management site(s) chosen for the program), the incident-free ground transport of foreign research reactor spent nuclear fuel in the United States is estimated to result in up to 0.30 (i.e., less than one) latent cancer fatalities over the entire duration of the program. This includes risk to both the public and the transportation workers. In other words, DOE and the Department of State would not expect any fatalities from cancer as a result of the ground transport of spent nuclear fuel if the proposed policy were implemented.

In the case of truck transport, truck driver(s) would be monitored for radiation dose. The regulatory limit of 100 mrem per year would never be reached during any single shipment, but the same driver could be used for multiple shipments throughout the year. DOE would implement mitigation measures through the foreign research reactor spent nuclear fuel acceptance contracts to ensure that each individual driver's dose remains below the regulatory limit. Should any individual truck driver's accumulated dose approach the 100 mrem limit in a year, DOE would require that a new driver(s) be used to keep each individual driver's dose below the regulatory limit.

S.4.2.3.2 Impacts of Accidents During Ground Transport

The most severe ground transport accidents would be truck or train crashes, followed by a large fire. Although this type of accident is highly unlikely, total ground transportation accident risks would be up to 0.00028 latent cancer fatalities from radiation and up to 0.14 for traffic fatalities depending on the transportation mode and foreign research reactor spent nuclear fuel management sites. The radiological risk of 0.00028 latent cancer fatalities means that the chance of any additional cancers among the population due to a ground transport accident is less than one in 1,000. The risk of 0.14 for a traffic fatality means that, under these conservative assumptions, there would be a 14 percent chance of a traffic fatality.

For the maximally exposed individual member of the public along the transportation route, the radiological risk from ground transport accidents would be 0.000000000014, or less than one chance in 10 billion of that individual incurring a fatal cancer.

The use of NRC- and Department of Transportation-approved routes and the development of specific foreign research reactor spent nuclear fuel transportation plans that would incorporate and integrate State and local emergency response plans would increase emergency responder effectiveness and reduce the potential consequences of a foreign research reactor spent nuclear fuel accident.

S.4.2.4 Foreign Research Reactor Spent Nuclear Fuel Management Site Impacts

The EIS examined the potential environmental impacts resulting from activities at the proposed management sites under the basic implementation of Management Alternative 1. The analysis examined environmental topics including land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational and public health and safety, noise, traffic and transportation, utilities and energy, and waste management. The analysis showed that at any of the proposed spent nuclear fuel management sites (the Savannah River Site, the Idaho National Engineering Laboratory, the Nevada Test Site, the Oak Ridge Reservation, and the Hanford Site), the potential impacts on the environment would be low. Further, there were no major differences among the spent nuclear fuel management sites for any of these environmental topics.

Potential radiation exposures to workers and the public at the management sites would be low. The EIS characterized the number and location of minority and low-income populations residing near candidate management sites. Minority or low-income populations living near the proposed management sites would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. Rather, they would be subjected to very low impacts as would the general population.

Implementation of the proposed action would have few nonradiological effects on the environment at management sites, including the social and economic status of the general population, minority populations, and the low-income population surrounding management sites. The EIS analyses show that the economic benefits resulting from increased cargo handling, transportation, and storage at management sites would be small for the general population or any particular segment of the population residing near management sites.

S.4.2.4.1 Impacts from Incident-Free Management Site Activities

The EIS analyses show that the risk to the maximally exposed individual member of the public from incident-free operations on DOE's spent nuclear fuel management sites would be 0.00000014 latent cancer fatalities for the duration of the foreign research reactor spent nuclear fuel receipt period. This hypothetical individual would be living at the site boundary of the Oak Ridge Reservation. This represents less than one chance in one million that this hypothetical individual would develop a latent cancer fatality due to the proposed spent nuclear fuel management activities.

The greatest population risk to the public living near any of DOE's spent nuclear fuel management sites would be 0.00027 latent cancer fatalities resulting from the combination of Phase 1 (i.e., near-term) receipt and storage of foreign research reactor spent nuclear fuel at the Savannah River Site and Phase 2 (i.e., future) receipt and storage at the Oak Ridge Reservation.

The highest estimated population risk to the workers who perform foreign research reactor spent nuclear fuel handling operations would be 0.21 latent cancer fatalities resulting from the combination of the near-term receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory, and the future receipt and storage at a different site. The maximum exposure to an individual worker was not calculated due to the large uncertainties involved with such calculations. A very conservative upper bound, however, would be the regulatory limit of 5,000 mrem per year, which translates to 0.026 latent cancer fatalities for workers receiving such a dose for the 13-year period during which receipts could take place.

S.4.2.4.2 Impacts of Accidents During Management Site Activities

The analysis of hypothetical accidental radioactive releases included meteorological conditions at the sites, population distributions, and food production and consumption rates within 80 kilometers (50 mi) of the storage location. Accident scenarios consisted of fuel assembly breach, dropped fuel cask, aircraft crash with and without fire, and accidental criticality. Consequences were estimated for a member of the public at the nearest site boundary and the population within 80 kilometers (50 mi) of the management site.

The highest estimated risk of incurring a latent cancer fatality for the maximally exposed individual member of the public would be 0.000010 for the duration of the foreign research reactor spent nuclear fuel receipt and storage period at the Oak Ridge Reservation. This represents one chance in 100,000 that this hypothetical individual would develop a latent cancer fatality. The greatest population risk to the public would be 0.11 latent cancer fatalities resulting from hypothetical accident conditions during Phase 1 receipt and storage at the Savannah River Site followed by Phase 2 receipt and storage at the Oak Ridge Reservation.

S.4.2.4.3 Other Potential Environmental Impacts from Management Site Activities

The EIS characterized each environmental component that would be impacted by site activities resulting from the basic implementation of Management Alternative 1.

Land Use. For all proposed management sites, unless all of the spent nuclear fuel is chemically separated or otherwise processed, new storage facilities for spent nuclear fuel including foreign research reactor spent nuclear fuel would be built on land previously disturbed or designated for industrial use. No additional land outside of the existing sites would be required for foreign research reactor spent nuclear fuel storage. The largest land use impact would be 16 ha (40 acres) at Oak Ridge Reservation to construct a new dry storage facility (less than 0.1 percent of the total site).

Socioeconomics. No construction personnel would be needed for existing facilities, and no more than 240 workers per year (peak) would be needed to build a new dry storage facility. Annual staffing requirements for operations would be about 30 full-time employees during receipt and 8 full-time employees during storage for a new dry storage facility. This would represent 0.15 to 0.9 percent of the existing work force at any of the proposed sites. No new hiring would be expected because most positions would be filled by reassignments of the existing work force. Even if all operational positions were filled by new hires, this would represent a small increase in regional employment.

Cultural Resources. Although most of the potential management sites contain areas of archaeological, cultural, or historic interest, little or no direct impacts on cultural resources would be expected because of the location of the storage facilities. However, site surveys would be conducted prior to construction. In the event that cultural resources were found, the State Historic Preservation Officer would be contacted. Tribal leaders would be contacted if any Native American resources were found.

Aesthetic and Scenic Resources. New storage facilities would be located far from public view in areas previously disturbed or designated for industrial use. Construction activities would generate dust that could temporarily affect visibility. Every effort would be made, however, to minimize such conditions. Facility operations would not produce emissions that would affect visibility.

Geology. Except for the potential existence of gold, tungsten, and molybdenum at Nevada Test Site, geologic resources consist of sand, gravel, or clay deposits, all of which have low economic value. Construction activities would disturb these surface deposits, but because of the large volume of these materials on the potential sites, the impact would be small.

Air Quality. Construction activities would cause temporary, minor increases in dust emissions, but the use of standard dust-suppression techniques would mitigate this problem. Overall, particulate emissions during construction could temporarily affect visibility in localized areas but would not exceed Federal or State requirements.

Water Quality. Water consumption during construction would require very small amounts of water when compared to daily water usage at the proposed management sites. During operations, the maximum annual water consumption would be about 2.1 million liters (550,000 gal). This amount represents no more than 0.2 percent of the annual water consumption at any of the proposed foreign research reactor spent nuclear fuel management sites. At the Nevada Test Site, where available water is limited, a cumulative water supply impact would be significant from activities other than foreign research reactor spent nuclear fuel management, but the foreign research reactor spent nuclear fuel management contribution would be very small. Under normal operations, there would be no direct discharge or effluent to ground or surface waters from a new dry storage facility.

Ecology. During construction of new facilities, individual or small populations of some wildlife species could be disturbed, displaced, or destroyed. However, the size of the affected areas would be small compared to the size of the remaining natural habitats.

Noise. Construction activities would generate noise levels consistent with light industrial activity. Based on existing studies, these noises would not be expected to propagate offsite at levels that would affect the general population. Noises generated during operations would be less than that during construction.

Materials, Utilities, and Energy. For existing facilities, incremental increases in materials, utilities, and energy would be very small. New dry storage facilities would result in increased demands on water, power, and sewage. Increased water usage during construction would add no more than 0.2 percent to existing site-wide levels. Increased annual electricity requirements would be about 800 to 1000 megawatt-hours per year. Increased sewage generation would be less than one percent above existing site-wide levels. At the Nevada Test Site, a central sewage system would have to be constructed for spent nuclear fuel management activities, including foreign research reactor spent nuclear fuel storage facilities. However, all other existing system capacities would manage the estimated increases for materials, utilities, and energy.

Waste Management. At all proposed foreign research reactor spent nuclear fuel management sites, the amount of waste generated from foreign research reactor spent nuclear fuel storage would be very small when compared to annual waste projection for each site, and could be handled by existing capacity at each site.

S.4.2.4.4 Cumulative Impacts at the Management Sites

The contribution to cumulative impacts from activities required for foreign research reactor spent nuclear fuel storage at any site would be very small in comparison with other spent nuclear fuel management activities and even smaller in comparison with other ongoing and reasonably expected non-spent nuclear fuel-related projects. A cumulative impact results from the incremental impact of a contemplated action added to the impacts of other past, present, and reasonably foreseeable future actions.

S.4.2.4.5 Impacts of Ultimate Disposition

Because title to the foreign research reactor spent nuclear fuel would pass to the United States if the proposed policy were adopted and foreign research reactor spent nuclear fuel were accepted into the United States, the Nuclear Waste Policy Act provides authority for its disposal in a geologic repository. A separate environmental evaluation of proposed geologic disposal activities would be conducted prior to such disposal.

It is possible that the foreign research reactor spent nuclear fuel could be accepted intact in a geologic repository. If DOE determines that geologic disposal of intact foreign research reactor spent nuclear fuel is possible, then there would be no onsite impacts beyond those associated with storage and packaging of the foreign research reactor spent nuclear fuel.

It is also possible that some form of processing (e.g., that associated with the new treatment technologies that would be examined under the preferred alternative) could be necessary to convert foreign research reactor spent nuclear fuel into a more stable form prior to its ultimate disposal. This processing could be a near-term new treatment technology, conventional chemical separation, or a new treatment technology that is implemented after an interim period of storage. The environmental impacts of such processing activities in the future cannot be precisely estimated at this time because the processes that might be used have not been fully developed. DOE expects that any new technology would produce no greater impacts than those that resulted from historical reprocessing activities in the United States. Therefore, the impacts of near-term treatment of the foreign research reactor spent nuclear fuel would be no greater than the impacts of chemically separating the same material as analyzed in the EIS. If a new treatment technology is implemented after an interim period of storage and technology development, then DOE expects that it would provide substantial improvements over conventional chemical separation.

When disposal space is available, DOE would transport the intact or processed foreign research reactor spent nuclear fuel to a repository. This transportation would be expected to produce impacts similar to the ground transportation impacts discussed in

Section S.4.2.3 of the Summary. After emplacement in a geologic repository, however, DOE expects there would be no more impacts to workers, the public, or the environment because the radioactive material would be effectively isolated.

In the event that the geologic repository were to be delayed, DOE assumed for the purposes of this analysis that it would continue to manage the foreign research reactor spent nuclear fuel, or the high-level radioactive waste form resulting from the chemical separation or other processing of such spent nuclear fuel, at the management sites until a geologic repository becomes available. The risk associated with this continued management is low and would not exceed the annual risk discussed in Section S.4.2.4.1.

S.4.3 Policy Considerations and Environmental Impacts from Implementation Alternatives of Management Alternative 1

In addition to the basic implementation of Management Alternative 1, the EIS analyzed implementation of Management Alternative 1 by various other means. The range of these implementation alternatives (which are variations on the basic implementation), deals with: 1) different amounts of material to be accepted; 2) different policy durations; 3) different financial arrangements; 4) alternative locations for taking title; 5) wet storage technology for new construction instead of new dry facilities; 6) near-term conventional chemical separation in the United States instead of interim storage in the United States; and 7) development and use of new treatment and/or packaging technologies instead of conventional chemical separation or storage. A discussion of the policy considerations and environmental impacts for each of the implementation alternatives follows. The impacts reported below cover the full range of activities (i.e., marine transport, port activities, ground transport, and site management activities) necessary to carry out the particular implementation alternative.

S.4.3.1 Implementation Alternative 1: Different Amounts of Material

The EIS evaluated impacts from accepting two different amounts of foreign research reactor spent nuclear fuel, plus target material, under this implementation alternative. These impacts are discussed below.

- *Implementation Subalternative 1a: Accept Foreign Research Reactor Spent Nuclear Fuel Only From Countries with Other-Than-High-Income-Economies*

  By excluding high-income economy countries, this subalternative would have adverse consequences for U.S. nuclear weapons nonproliferation policy. The amount of HEU that could be removed from international commerce under this implementation subalternative is less than ten percent of the amount that could be removed under the basic implementation. Furthermore, if this was the only spent nuclear fuel accepted, research reactor operators in high-income economy countries would be likely to implement several measures contrary to U.S. nuclear weapons nonproliferation policy, such as delaying or canceling plans to convert to LEU fuel, or, in some cases, reconverting from LEU to HEU fuel. The environmental impacts would be reduced in comparison with the basic implementation in direct proportion to the reduced amount of spent nuclear fuel accepted.

- *Implementation Subalternative 1b: Accept Only HEU Foreign Research Reactor Spent Nuclear Fuel*

  Foreign research reactor operators have stated that they would not participate in the Reduced Enrichment for Research and Test Reactors Program unless the United States accepts their spent nuclear fuel, including LEU spent nuclear fuel. Thus, this implementation subalternative could result in the end of that program. Furthermore, this implementation subalternative would be contrary to the broader U.S. nuclear weapons nonproliferation policy. Since the number of elements in this implementation subalternative is about half the number of elements in the basic implementation, the potential environmental impacts would be approximately half of those calculated for the basic implementation.

- *Implementation Subalternative 1c: Accept HEU and LEU Target Material in Addition to Foreign Research Reactor Spent Nuclear Fuel*

  This implementation subalternative would remove the most HEU from civil commerce and provides the most support to U.S. nuclear weapons nonproliferation policy. Acceptance of this material in addition to the spent nuclear fuel would give incentives to reactor operators producing radioisotopes to switch from HEU targets to LEU targets, thus removing additional HEU from future international civil commerce. As with the basic implementation, acceptance of this additional material would have a small impact on all environmental, health, and safety issues. The dose rate from casks loaded with target material would be lower than the dose rate from casks loaded with foreign research reactor spent nuclear fuel. Up to 140 additional cask shipments are estimated to be needed for this material. These cask shipments would include up to 125 overland Canadian shipments. The environmental impacts are expected to be slightly higher than those associated with the basic implementation due to these additional cask shipments. The total incident-free population risk to the exposed public and workers would be 0.58 latent cancer fatalities as compared with 0.55 latent cancer fatalities under the basic implementation of Management Alternative 1.

S.4.3.2 Implementation Alternative 2: Alternative Policy Durations

The EIS evaluates the impacts of reducing the policy duration to 5 years of spent nuclear fuel acceptance or of continuing the policy for acceptance of HEU spent nuclear fuel indefinitely and LEU for 10 years.

- *Implementation Subalternative 2a: 5-Year Policy*

  The amount of HEU that could be removed from international commerce under this implementation subalternative is about 88 percent of the amount that could be removed under the basic implementation. The 5-year policy would accelerate the point at which the foreign research reactor operators and governments would become responsible for disposal of their own spent nuclear fuel. This may not be enough time for some countries, especially other-than-high-income economy countries, to make arrangements for alternative means of managing their spent nuclear fuel.

Under this implementation subalternative, approximately 81 percent of the total number of shipments under the basic implementation would be needed. The environmental impacts under this implementation subalternative would be reduced as compared with the basic implementation in direct proportion to the lesser amounts of foreign research reactor spent nuclear fuel accepted. As in the basic implementation, the effects would be small and no fatalities from cancer or accidents would be expected.

- *Implementation Subalternative 2b: Indefinite HEU/10-Year LEU Policy*

  The amount of spent nuclear fuel would be the same as in the basic implementation — only the timing for shipment of the HEU spent nuclear fuel would be different. Indefinite acceptance of HEU would promote U.S. nuclear weapons nonproliferation goals by allowing more time to remove the HEU from international commerce. The potential environmental impacts would be the same as or slightly lower than those of the basic implementation. Delaying the acceptance of a small fraction of the total amount of foreign research reactor spent nuclear fuel accepted would have a very small effect.

S.4.3.3 Implementation Alternative 3: Alternative Financing Arrangements

The EIS evaluated three alternative financing arrangements. These are: 1) subsidize all countries; 2) charge all countries the full cost of accepting and managing their spent nuclear fuel; and 3) subsidize other-than-high-income economy countries and charge high-income economy countries the full cost of managing their spent nuclear fuel. The first financing arrangement would be the most expensive for the United States, while the second would cost the United States nothing, and the third would fall somewhere in between.

These financing arrangements could have an indirect effect on the environmental impacts of accepting foreign research reactor spent nuclear fuel because the number of foreign research reactor operators participating in the program would depend on the fee the United States proposed to charge. The indirect effects are impossible to quantify, but would only result in a reduction in the amount of HEU removed from international commerce and in the environmental impacts on United States territory.

S.4.3.4 Implementation Alternative 4: Alternative Locations for Taking Title

The EIS evaluated alternative locations for taking title. These include: prior to shipment; at the port(s) of entry; and at the proposed foreign research reactor spent nuclear fuel management site(s).

The environmental impacts of the proposed foreign research reactor spent nuclear fuel program are not affected by who the owner of the spent nuclear fuel is or the point at which title is transferred. The Price-Anderson Act would apply to the spent nuclear fuel shipments, once they arrive in territorial United States, regardless of who holds title to the fuel. Thus, there would be no change in the liability protection provided to the citizens of the United States, no matter where DOE would take title. Ownership would not affect shipping arrangements and precautions, or liability protection, other than to increase DOE's potential liability if DOE were to take title before shipment.

S.4.3.5 Implementation Alternative 5: Wet Storage Technology for New Construction

Wet storage technology was evaluated under the EIS as a site storage option for Phase 2 storage. At the conclusion of Phase 1, the spent nuclear fuel could be stored in new wet storage facilities or in one non-DOE facility (Barnwell Nuclear Fuels Plant) located adjacent to the Savannah River Site which could be acquired and refurbished for use as a wet storage facility. This implementation alternative would support U.S. nuclear weapons nonproliferation policy to the same extent as the basic implementation.

Impacts to the health and safety of the public and workers would be similar to those discussed for new dry storage in the basic implementation. The risk of an accidental criticality, however, is higher for wet storage technology than for dry storage technology. Thus, the total population risk to the public due to accident conditions would be 0.16 latent cancer fatalities under this implementation alternative, compared to 0.11 latent cancer fatalities under the basic implementation.

The highest maximally exposed individual risk to the public due to accident conditions would be 0.00015 latent cancer fatalities under this implementation alternative, which is the highest of all the alternatives. This individual's chance of incurring a latent cancer fatality would be less than two in 10,000.

S.4.3.6 Implementation Alternative 6: Near-Term Conventional Chemical Separation in the United States

The EIS evaluates near-term conventional chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory for five key environmental impacts; 1) waste management; 2) air quality; 3) water quality; 4) occupational and public health and safety; and 5) socioeconomics. The facilities at the Savannah River Site are technically capable of chemically separating the aluminum-based foreign research reactor spent nuclear fuel. After some upgrading, the facilities at the Idaho National Engineering Laboratory would be technically capable of chemically separating all the foreign research reactor spent nuclear fuel.

The same amount of HEU could be removed from international commerce under this implementation alternative as under the basic implementation. Foreign research reactor operators would have the same incentives not to use HEU in their reactors under this implementation alternative as they would under the basic implementation.

The principal environmental impacts under this implementation alternative would be occupational and public health and safety impacts. The total incident-free population risk to the worker population of incurring a latent cancer fatality resulting from this implementation alternative would be 0.32 latent cancer fatalities among all marine, port, ground transport, and site workers combined. The largest contribution to this total risk would be from onsite radiation workers. The risk to onsite radiation workers would be 0.21 latent cancer fatalities, which translates to 21 chances in 100 that one worker within the group of exposed workers would develop a latent cancer fatality.

The total incident-free population risk to the general public would be 0.39 latent cancer fatalities among the entire affected population under this implementation alternative. The total population risk due to accident conditions to the general public would be 0.43 latent cancer fatalities among people living near the affected site.

S.4.3.7 Implementation Alternative 7: Developmental Treatment and/or Packaging Technologies

This implementation alternative could be selected in connection with other implementation alternatives. The environmental impacts of the developmental treatment and/or packaging technologies cannot be precisely estimated at this time because the technologies and procedures are still under development. Implementation of certain treatment and/or packaging technologies would require new facilities and thus would generate impacts associated with construction as well as operation. Appropriate NEPA documentation would be prepared for any proposed implementation of new treatment and/or packaging technologies. A new facility using a new treatment technology would not be operational in the near-term, so in this case, this implementation alternative would be selected in conjunction with one of the near-term storage alternatives.

Any new facilities would be designed to meet modern environmental compliance and health and safety standards. The new design would minimize air and water emissions and would limit the public and worker radiation doses to levels no greater than those in existing facilities. Therefore, it is expected that these impacts would be somewhat lower than those presented for conventional chemical separations.

S.4.4 Implementation of the Preferred Alternative

Under the preferred alternative, as described in Section S.2.3, DOE would accept and manage the foreign research reactor spent nuclear fuel and target material in the United States. The aluminum-based foreign research reactor spent nuclear fuel and target material would be transported to and managed at the Savannah River Site. The TRIGA foreign research reactor spent nuclear fuel would be transported to and managed at the Idaho National Engineering Laboratory. Under the preferred alternative, up to 17,800 aluminum-based foreign research reactor spent nuclear fuel elements representing approximately 675 casks, and the target material from overseas, would arrive at candidate ports on the east coast of the United States, preferably the Naval Weapons Station at Charleston, South Carolina. Most of the target material would be received at the U.S.-Canadian border and all target material, representing 140 casks, would be managed at the Savannah River Site. Up to approximately 38 casks of TRIGA foreign research reactor spent nuclear fuel could arrive at candidate ports on the United States west coast, preferably the Naval Weapons Station Concord, California. DOE would strive to minimize the number of shipments necessary by coordinating shipments from several reactors at a time (i.e., by placing multiple casks [up to eight] on a ship). DOE currently estimates that approximately five shipments through the Naval Weapons Station Concord would be necessary. All the TRIGA foreign research reactor spent nuclear fuel, representing approximately 162 casks and 4,900 elements would be transported to and managed at the Idaho National Engineering Laboratory.

The policy considerations and the impacts of marine transport, port, ground transport, and management site activities of the preferred alternative presented in this section are based on analysis performed for the basic implementation of Management Alternative 1 (Section S.4.2), Implementation Alternative 1c (Section S.4.3.1), Implementation Alternative 6 (Section S.4.3.6), and Implementation Alternative 7 (Section S.4.3.7).

S.4.4.1 Policy Considerations

A critical result of implementing the preferred alternative would be support for the Reduced Enrichment for Research and Test Reactors Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs. The successful expansion of the program to Russia, other Newly-Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU is dependent on the United States commitment to action such as that embodied in this preferred alternative. By including the target material, the preferred alternative maximizes the amount of HEU to be removed from international commerce. By assisting foreign research reactor operators with peaceful applications of nuclear energy, the preferred alternative complies with U.S. obligations under the *Treaty on the Non-Proliferation of Nuclear Weapons*. By not encouraging reprocessing for either nuclear power or nuclear explosive purposes, the preferred alternative supports the Administration's nuclear weapons nonproliferation policy objectives.

DOE's preferred alternative allows for the use of chemical separation under certain circumstances, such as when alternative technologies present higher safety risks, are more costly, or are unavailable. If chemical separation is used to process the foreign research reactor spent nuclear fuel, the HEU would be blended down during the separation process to a low enriched form that is unsuitable for nuclear weapons purposes (the blenddown is also required because the F-canyon cannot safely process HEU beyond initial dissolution). No plutonium would be separated. Instead, the plutonium would be left in the waste stream with the high-level radioactive chemical separation wastes. In addition, the waste generated during reprocessing would be handled using technologies that are intended to be used for substantially larger quantities of pre-existing wastes (e.g., vitrification of high-level liquid radioactive wastes, grouting for low-level wastes, and incineration for some supernatant).

This potential method of handling the foreign research reactor spent nuclear fuel would be consistent with United States nonproliferation policy, despite the use of chemical separation because (1) it would reduce the worldwide stockpiles of this nuclear weapons material; (2) no plutonium would be separated; and (3) the chemical separation would not be taking place for either nuclear weapons or nuclear power purposes.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These conditions will be addressed by the independent study described in S.2.3.

S.4.4.2 Potential Environmental Impacts

The potential environmental impacts due to marine transport, port activities, and ground transport would be similar to those of the basic implementation of Management Alternative 1, with the addition of the target material shipments. With the use of only military ports, which are located in areas of low population density, the risk would be reduced for port activities.

Management site activities at the Idaho National Engineering Laboratory can be estimated from the basic implementation of Management Alternative 1. For the Savannah River Site, however, the impacts would vary depending on the specific outcome of the preferred management strategy at the site. The preferred alternative includes the development and operation of a new treatment and/or packaging technology at the Savannah River Site, and environmental impacts of such a technology cannot be estimated with precision. DOE expects, however, that the radiological and nonradiological health and environmental effects from the operation of facilities that would support a new technology would not exceed those estimated for conventional chemical separation (evaluated in S.4.3.6).

The principal impacts of the preferred alternative would be occupational and public health and safety impacts. Radiological risks are determined for the maximally exposed individual and the potentially exposed population. The maximally exposed individual risk expresses the probability that the maximally exposed individual would incur a latent cancer fatality due to the preferred alternative. The population risk expresses the estimated number of additional latent cancer fatalities among the entire potentially exposed population. Risks are also determined for incident-free conditions and for accident conditions.

The greatest radiological risks would occur during ground transport or management site activities. Based on conservative assumptions, under incident-free conditions, the total population risk would be 0.25 latent cancer fatalities for the potentially exposed public, while the corresponding risk would be 0.30 latent cancer fatalities for the workers. Thus, there would be an estimated 25 percent chance of incurring one additional latent cancer fatality among the exposed general public, and a 30 percent chance of incurring one additional latent cancer fatality among workers. The maximum estimated incident-free radiological risk for an onsite radiation worker would be 0.026 latent cancer fatalities or a 2.6 percent chance of incurring a latent cancer fatality. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation.

Under accident conditions, the maximum population risk to the general public (which would be to the people living near both management sites at the time of an accident) would be 0.45, or an approximate 45 percent chance of incurring one additional latent cancer fatality among all the people living near both sites. The maximum estimated accident radiological risk to the maximally exposed individual is 0.000047 latent cancer fatalities, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring a latent cancer fatality due to an accident under this alternative would be less than one in 10,000. There is approximately a five percent chance that a truck driver or member of the public could die in a traffic accident associated with the preferred alternative. This death would be unrelated to the radioactive nature of the cargo.

The cumulative impacts from the receipt and management of foreign research reactor spent nuclear fuel and target material, with some chemical separation, would not have detrimental effects on the public or environmental resources at the sites. Minority and low-income populations living near the Savannah River Site or the Idaho National Engineering Laboratory would not be subjected to any disproportionately high and adverse impacts.

S.4.5 Policy Considerations and Environmental Impacts of Management Alternative 2

Under Management Alternative 2 the United States would facilitate overseas management of foreign research reactor spent nuclear fuel. Two implementation subalternatives are examined: 1) overseas storage of the foreign research reactor spent nuclear fuel with U.S. assistance; and 2) overseas reprocessing of the foreign research reactor spent nuclear fuel with U.S. nontechnical (financial and/or logistical) assistance. Under these implementation subalternatives, no foreign research reactor spent nuclear fuel would be accepted into the United States. However, vitrified waste from overseas reprocessing might be accepted into the United States resulting in some environmental impacts. In addition, other impacts would occur, such as cost of U.S. assistance and impacts to the U.S. nuclear weapons nonproliferation policy.

S.4.5.1 Impacts From Overseas Storage With U.S. Assistance

Under this subalternative, there would be no environmental impacts on U.S. territory for the duration of the proposed action. However, other impacts would occur such as cost of U.S. assistance. This subalternative would be most economical in countries that already have spent nuclear fuel storage infrastructures. In these countries, the addition of the spent nuclear fuel from research reactors to existing spent nuclear fuel inventories in storage would probably involve incremental costs without start-up costs. However, in countries that do not have an existing nuclear infrastructure, it would cost tens of millions of dollars per country to set up a secure area, purchase storage casks, transfer the spent nuclear fuel to the cask, and maintain a secure area for 40 years.

If the United States does not accept any near-term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risk would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

S.4.5.2 Impacts From Overseas Reprocessing With U.S. Nontechnical Assistance

The EIS considers a subalternative in which all of the reprocessing activities occur overseas using foreign reprocessing and vitrification technology. In one subalternative, the vitrified high-level radioactive waste (i.e., high-level waste would be incorporated into high strength, dissolution-resistant glass and cast in stainless steel canisters) from overseas reprocessing would be accepted by the United States for storage and/or ultimate disposal. If no high-level waste were accepted, then there would be no impacts on U.S. territory.

In the case where vitrified high-level waste from overseas reprocessing is accepted into the United States from Europe, the environmental impacts in the United States would be, in all cases, lower than those under the basic implementation. The total number of casks received in the United States would be 90-95 percent lower and the total worker and population exposures would be lower. If vitrified high-level waste is accepted by the United States, it would either be stored in existing facilities at the Savannah River Site or shipped directly to a geologic repository site, if available.

S.4.6 Policy Considerations and Environmental Impacts of Management Alternative 3

Implementation of this Management Alternative 3, a combination of elements from Management Alternative 1 and 2, would not pose a greater risk than that determined under Management Alternative 1, assuming identical United States site management technology implementation. This is because the amount of foreign research reactor spent nuclear fuel that would be managed under the Hybrid Alternative is less than the amount that would be managed in the United States under Management Alternative 1.

S.4.7 No Action Alternative

If no action were taken to adopt a policy to manage foreign research reactor spent nuclear fuel, no direct environmental impacts would occur in the United States. However, failure to adopt a policy would have numerous other impacts, including likely continued reliance on HEU by foreign research reactor operators, increasing the amount of HEU available in civilian commerce, and possible worldwide proliferation of nuclear weapons and nuclear weapons States. The No Action alternative would be the least desirable from a nuclear weapons nonproliferation standpoint.

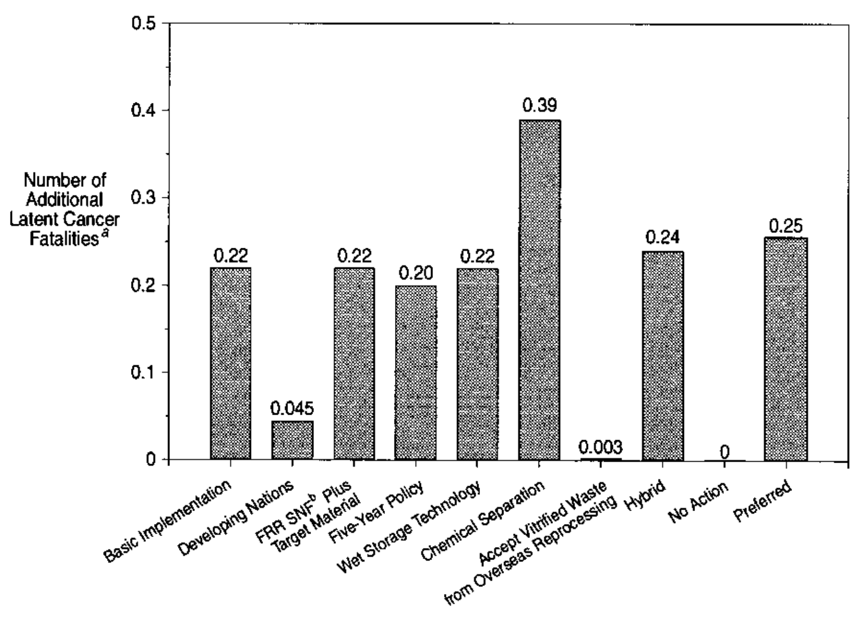
S.4.8 Comparison of the Radiological Risks

This section provides a comparison of the potential maximum estimated risks associated with the alternatives evaluated in the EIS for the general public, the workers, and the maximally exposed individual. Essentially this risk would occur during the first 13 years of the program.

Figure S-11 shows the greatest incident-free population risk to the general public under each alternative. Figure S-12 shows the greatest accident population risk to the general public under each alternative. These estimated risks (including the maximum estimated risk of 0.39 latent cancer fatalities under incident-free conditions, and 0.45 latent cancer fatalities under accident conditions) would be less than one-half additional latent cancer fatality among the public living near [within 80 kilometers (50 mi)] any of the management sites.

The accident risks to the population are estimated by combining the probabilities of accidents and the consequences of those accidents, then summing over the full range of accidents that might reasonably be expected to occur during marine transport, port activities, ground or barge transport, and management site activities. The single accident with the highest risk is estimated to have a probability of approximately 0.02 occurrences per year and a consequence of approximately 1.3 latent cancer fatalities.

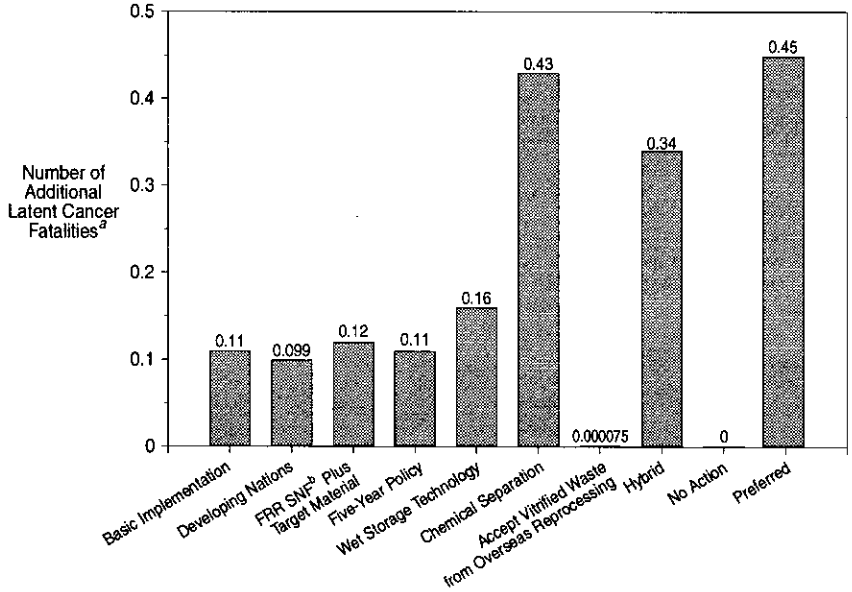
Incident-free population risks for workers are depicted in Figure S-13. The greatest incident-free radiological population risk to workers from any of the alternatives would occur in the alternative in which target material is added to the basic implementation of



[a] Impacts evaluated were those in the United States and on the Global Commons.

[b] Foreign Research Reactor Spent Nuclear Fuel

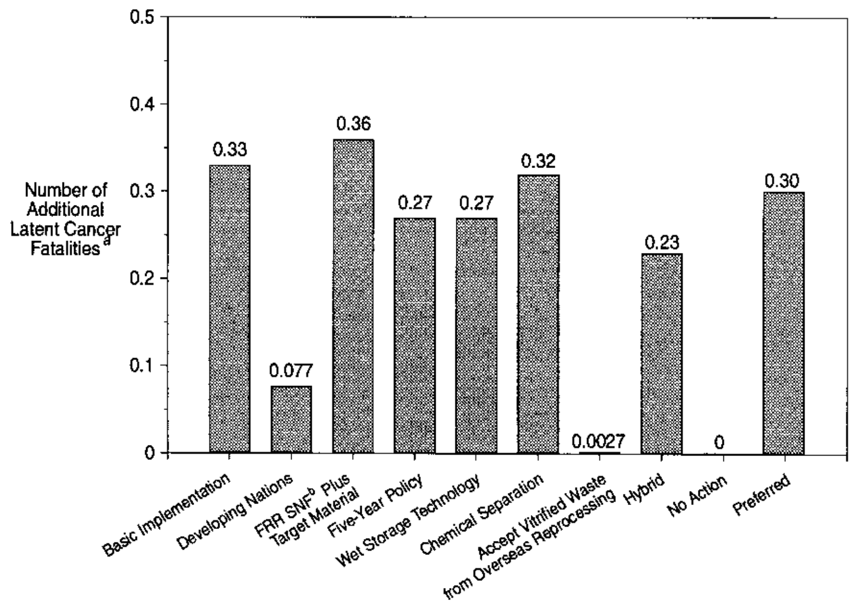
Figure S-11 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative



[a] Impacts evaluated were those in the United States and on the Global Commons.

[b] Foreign Research Reactor Spent Nuclear Fuel

Figure S-12 Maximum Estimated Accident Radiological Population Risk to the General Public Under Each Alternative



[a] Impacts evaluated were those in the United States and on the Global Commons.

[b] Foreign Research Reactor Spent Nuclear Fuel

Figure S-13 Maximum Estimated Incident-Free Radiological Population Risk for Workers Under Each Alternative

Management Alternative 1. This low risk would mainly occur during the first 13 years of the program and could be up to 0.36 additional latent cancer fatalities to the worker group under incident-free conditions.

The analysis in this EIS indicates that the highest estimated individual risk would be to an onsite radiation worker (i.e. the maximally exposed worker) receiving a dose equal to the regulatory limit of 5,000 mrem each year for 13 years under incident-free conditions. This risk would be 0.026 latent cancer fatalities, and this hypothetical individual's estimated chance of developing a fatal cancer would be 2.6 chances in 100. However, DOE would prevent workers from receiving this high radiation dose through existing administrative procedures.

The EIS analysis indicates that the highest estimated maximally exposed individual risk to members of the public under the proposed action is 0.00015 latent cancer fatalities. This would be a hypothetical individual member of the public who was at the worst possible location during an accidental criticality on the Oak Ridge Reservation under Implementation Alternative 5, Wet Storage Technology for New Construction. This accident is estimated to have a frequency of approximately 0.0031 occurrences per year and a consequence of approximately 0.0017 latent cancer fatalities. This hypothetical individual's chance of incurring a fatal cancer would be increased by less than two in 10,000.

The highest estimated incident-free population risk to the general public (See Figure S-11) living near any of the management sites, or ports, from any of the implementation alternatives is less than one-half latent cancer fatality. This risk occurs under Implementation Alternative 6, near-term Chemical Separation in the United States at the Savannah River Site. This risk would be spread among the roughly 600,000 people who live near the Savannah River Site, so the average risk among these people would be less than one in a million over the entire 40-year period. Common activities that produce a comparable risk of death per year are presented in Table S-1.

Table S-1 Risks Estimated to Increase Chance of Death in Any Year by One Chance in a Million

Activity	Cause of Death
Smoking 1.4 cigarettes	Cancer; Heart disease
Living 2 days in New York or Boston	Air pollution
Traveling 16 km (10 mi) by bicycle	Accident
Flying 1,600 km (1,000 mi) by jet	Accident; Cancer caused by cosmic radiation
Living 2 months in Denver on vacation from New York	Cancer caused by cosmic radiation
One chest x-ray taken in a good hospital	Cancer caused by radiation
Drinking 30 12-oz cans of diet soda	Cancer caused by saccharin

Adapted from Slovic, P., 1986, Informing and Educating the Public about Risk, Risk Analysis, Vol. 6(4), pp. 403-415.

S.4.9 Costs

The program costs of implementing the proposed action alternatives described in Section S.2.1 and the preferred alternative are given in Table S-2. As noted in Section S.2.3, the preferred alternative is Management Alternative 1 with modifications to several

Table S-2 Potential Total Costs
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	Minimum Program Cost	Other Cost Factors (Technical)	Other Cost Factors (Discount Rate)	Potential Total Cost, No Escalation	1% Real Escalation, Cumulative
Management Alternative 1 (Storage)[b]	725/775[a]	210	175	~1,100	+11%
Management Alternative 1 (revised to incorporate Chemical Separation)[b]	625	85-145	125	~900	+9%
Management Alternative 2[b]	1250	600-1600	250	2,100-3,100	+13%
Management Alternative 3[b,c]	675	225-275	75	~1000	+9%
Preferred Alternative	625-950	210	225	~1,050-1,400	+10%-11%

a *Dry/Wet New Storage Facilities*
b *Adding target material to any of these scenarios would increase scenario costs by 3 to 4 percent*
c *The total cost risk to the United States is less than 1/2 the total cost risk since a large portion of the activities under this alternative would occur overseas*

basic implementation components, including implementation of a new treatment and/or packaging technology at the Savannah River Site for the management of the aluminum-based foreign research reactor spent nuclear fuel. A more detailed cost discussion can be found in Section 4.9 of Volume 1 and in Appendix F of the Final EIS. Costs are generally presented as minimum costs for the defined program components and approaches, and as "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum costs. However, for the preferred alternative, a wide range of costs is presented because of the uncertainty associated with the new technology development program. An example of an item covered by "other cost factors" would be the cost growth caused by adverse weather that extends the time required to make shipments of the spent nuclear fuel. All costs are discounted (i.e. the costs reflect long-term interest that can be expected) at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States.

Table S-2 combines the minimum costs with the other cost factors and shows the net present value of the potential total costs of implementing the program. Costs to manage target material are included in the preferred alternative. The total costs range from about $900 million for Management Alternative 1 with chemical separation, $1.4 billion for the preferred alternative to $3.1 billion for Management Alternative 2. Except for Management Alternative 2, overseas management, the minimum costs for other alternatives are $625-$950 million.

Costs to the United States have also been calculated in each of the cost scenarios under a variety of fee schedules. DOE would establish any fee in a separate *Federal Register* notice subsequent to the issuance of this EIS. Costs to the United States under the preferred alternative would be $275-550 million for a fee of $2,000/kgTM. Higher fees would reduce this cost to zero in the $7,500/kgTM range.

S.5 Overview of the Public Comments and DOE Response

On April 21, 1995, DOE published in the *Federal Register* a Notice of Availability of the *Draft Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel* (60 FR 19899). In accordance with DOE NEPA regulations 10 CFR Part 1021, the Notice invited interested agencies, organizations, and the general public to provide oral and written comments on the Draft EIS.

S.5.1 The Public Comment Process

The public comment period on the Draft EIS was initially scheduled from April 21, 1995 to June 20, 1995. In response to public requests, the comment period was extended an additional 30 days through July 20, 1995. During the comment period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and five candidate management sites. In addition, a public hearing was held in Washington, D.C. The hearing dates and locations are shown in Figure S-14. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

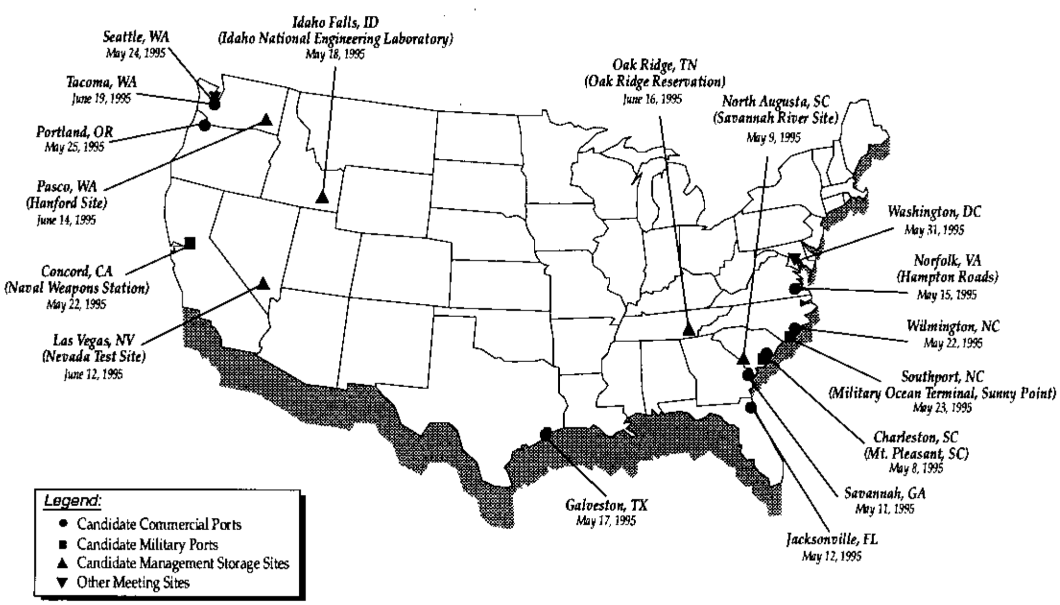


Figure S-14 Public Comment Hearing Locations and Dates

S.5.2 Written Comments

DOE received approximately 5,040 written comments contained within approximately 1,250 submissions. Written comments were submitted to DOE by mail and facsimile and at many of the public hearings. These written comments were received from individuals, Federal and State agencies, Tribal governments, local governments, foreign entities, and non-government organizations such as environmental, public interest, and industry groups. All written comments were reviewed and considered in the preparation of the Final EIS and are presented in Section 2 of Volume 3 of the Final EIS.

S.5.3 Public Hearings

In an effort to encourage a dialogue between members of the public and government officials at the public hearings, DOE used an informal, interactive format and an independent professional facilitator. The hearings were preceded by an hour-long "open house" at which exhibits, videos, and other information materials were available for review, along with opportunity for one-on-one exchanges with DOE representatives. Comment forms were provided for those wishing to submit written comments at the hearings.

Public hearings began with an explanation of the hearing format by the independent facilitator, followed by a 30-minute overview by a DOE official on the proposed policy and the factors leading to the proposal's development. Following this presentation, attendees were encouraged to ask questions, offer comments, and engage in dialogue. Notetakers summarized the questions and comments and DOE responses at all hearings. A summary of all oral comments and statements from each hearing, along with the DOE responses, is presented in Volume 3, Section 3 of the EIS.

Approximately 900 people attended the 17 public hearings. An interactive format was used at all hearings except in Tacoma, Washington. At the Tacoma public hearing, attendees expressed a desire for a more traditional approach in which people presented statements of up to five minutes, with little or no dialogue between commentors and DOE. In addition, the Tacoma hearing attendees requested that a verbatim transcript be made of the meeting. A copy of this transcript is included as Attachment 1 to Volume 3, Section 3 of the EIS.

S.5.4 Environmental Protection Agency Rating of EIS

The U.S. Environmental Protection Agency reviewed and rated the Draft EIS proposed action and each alternative as "lack of objections (LO)," which means that the EPA has not identified any potential environmental impacts requiring modifications to the proposal. A copy of the U.S. Environmental Protection Agency rating is included among the written comments in Volume 3, Section 2 of the Final EIS.

S.5.5 Major Issues Raised by Commentors

The public comments addressed a wide range of policy, economic, and technical issues. Of the approximately 6,000 written and oral comments received, few were critical of, or directed against, the analytical methods presented in the Draft EIS. The following is a summary characterizing the most frequently raised issues and the corresponding summary of DOE's responses. (In each case, a summary of DOE's response is provided in bold text following the summary of the public comment.) DOE's full response to each specific comment and issue are provided in Sections 2 and 3, Volume 3 of the Final EIS.

S.5.5.1 Policy Considerations and Management Alternatives

Numerous comments and questions were received concerning the need for a policy to manage foreign research reactor spent nuclear fuel. Commentors questioned the need to adopt a policy to manage spent nuclear fuel from allied countries or from countries that are considered sufficiently developed to manage their own spent nuclear fuel. Other commentors questioned the objectives of the stated U.S. nuclear weapons nonproliferation policy and the rationale for considering the proposed policy, pointing out that some of the allied nations under the proposed action do not pose a nuclear weapons proliferation risk. ***The purpose of the proposed action is to support a U.S. nuclear weapons nonproliferation policy that seeks to reduce, and eventually eliminate, the use of highly-enriched (nuclear weapons-grade) uranium in civil programs worldwide. It is necessary to deal with spent nuclear fuel from the developed countries for several reasons.***

First, if the United States does not assist the developed countries with management of their spent nuclear fuel, the only mechanism available to them for spent nuclear fuel disposition would be to stay on or reconvert to use of highly-enriched uranium for fuel. Those who can accept the reprocessing wastes would disposition their spent nuclear fuel by having it reprocessed, and would recycle the remaining highly-enriched uranium. They would have to seek out sources of new highly-enriched uranium to make up for that burned, and to keep the enrichment level of the recycled uranium high enough to be of use. Since the United States could not ship additional highly-enriched uranium to them, they would likely resort to Russia or China as suppliers. Such actions could destroy all the progress made by the Reduced Enrichment for Research and Test Reactors program in attempting to eliminate the use of highly-enriched uranium in civil programs.

Second, many developed countries manufacture research reactors and sell them to developing countries. If, due to inaction by the United States, research reactors in the developed countries refuse to convert to low enriched uranium fuel, or switch back to the use of highly-enriched uranium fuel, their customers in developing countries would likely insist on obtaining reactors that also use highly-enriched uranium fuel.

Third, inaction by the United States that leads research reactors in developed countries to shut down due to the absence of a timely means of dispositioning of their spent fuel is likely to lead, rightly or wrongly, to accusations that the United States is failing to comply with its obligations under the Treaty on the Non-Proliferation of Nuclear Weapons to assist nonnuclear weapons States with peaceful applications of nuclear energy.

Some commentors further contended that the nuclear weapons nonproliferation objectives do not apply to the foreign research reactor spent nuclear fuel that contains low enriched uranium, which is not weapons-grade material. ***While it is true that low enriched uranium is not weapons-grade material, it is included in the policy because acceptance of low enriched uranium fuel would provide incentive for foreign research reactor operators to convert from highly-enriched uranium fuel to low enriched uranium fuel use. This incentive would be necessary to offset the considerable expense of conversion and the reductions in reactor***

capabilities and increased operating costs that generally accompany conversion to low enriched uranium fuel. Furthermore, by not accepting low enriched uranium, the United States would be penalizing the reactors that converted earlier under the Reduced Enrichment for Research and Test Reactors Program, because those reactors are now generating only low enriched uranium spent nuclear fuel. Since there is currently no alternative available for disposition of low enriched uranium spent nuclear fuel, the reactors that supported the Reduced Enrichment for Research and Test Reactors Program would be the first to shut down.

Some commentors believe that the continued production and export of nuclear materials by the United States appears to be in conflict with the stated U.S. nuclear weapons nonproliferation objectives. Several stated their opposition to the continued sale of new fuel for foreign research reactors, and a small number suggested that the United States cease production of all nuclear materials and develop alternative energy sources. ***As a result of the passage of the Energy Policy Act of 1992, the United States is prohibited from selling highly-enriched uranium to foreign countries, except under special conditions. Since enactment of this prohibition, no new licenses for the export of highly-enriched uranium have been issued by the United States. The United States is continuing to sell low enriched uranium since it is not a nuclear weapons material. Furthermore, by making low enriched uranium available, the United States is providing further support for reactor operators who agree to convert from use of highly-enriched uranium. With respect to development of alternative energy sources, DOE currently has on-going programs that are seeking to develop and promote use of various alternative energy sources, such as wind, water, and solar.***

Representatives of foreign research reactor operators enumerated several reasons why the United States should accept and manage foreign research reactor spent nuclear fuel. This generally held position was the result of the expiration of the Off-Site Fuels Policy, under which the United States had accepted foreign research reactor spent nuclear fuel. As a result, many foreign research reactor operators are running out of storage space for their spent nuclear fuel. The operators assert that this may cause some research reactors to shut down, as these countries do not have, and did not plan for, long-term storage facilities. Foreign research reactor operators point out that the Dounreay, United Kingdom, reprocessing facilities can only handle highly-enriched uranium at this time. As a result, foreign research reactor operators would likely revert to reliance on highly-enriched uranium which would be contrary to U.S. nuclear weapons nonproliferation policy.

Many members of the public and State and local governments supported the objectives of the proposed action, but urged further consideration of Management Alternative 2, which is to facilitate overseas management of the spent nuclear fuel with security precautions to ensure that the spent nuclear fuel is not diverted into a nuclear weapons program. Some also supported the No Action Alternative under which the United States would neither accept nor assist with the management of foreign research reactor spent nuclear fuel. Several Non-Government Organizations also expressed support for the proposed policy and cited the need to eliminate the use and stockpiling of highly-enriched uranium. ***DOE and the Department of State have considered these comments in selection of the preferred alternative. DOE's and the Department of State's reasons for not selecting the No Action Alternative or Management Alternative 2 are discussed in Section S.2.3 under the heading "Basis for the Preferred Alternative."***

Many commentors expressed concern about the cost to the United States of managing foreign research reactor spent nuclear fuel. Several opposed full subsidization of developed countries which they consider capable of managing their own spent nuclear fuel. Other commentors favored competitive pricing or charging the foreign research reactor operators a full-cost recovery fee for management of their spent nuclear fuel. Representatives of certain foreign research reactor operators expressed their willingness to pay a cost-based price, and stated that they are not asking U.S. taxpayers to subsidize their fuel cycles. A number of commentors asked for additional information in the Final EIS on life cycle costs, risks, and benefits. ***DOE and the Department of State have evaluated several financing options in the EIS, ranging from fees from the research reactor operators that would pay all of the costs of the program to full subsidization of the program by DOE. One of these options would be for developed countries (which represent about 87 percent of the spent nuclear fuel total mass and about 78 percent of the spent nuclear fuel elements) to pay a competitive fee for U.S. management of their spent nuclear fuel. As part of this option, DOE would subsidize the costs of managing the spent nuclear fuel from developing countries. The United States does not believe the developing countries can afford to pay the expense for spent nuclear fuel management either in the United States or in the host country.***

S.5.5.2 Ultimate Disposition

The ultimate disposition of DOE-owned spent nuclear fuel was a widely expressed policy concern. Many commentors, concerned with a lack of long-term storage options, raised the issue of the availability, or lack thereof, of a permanent geologic repository. Many urged that, before the United States accepts any spent nuclear fuel from foreign research reactors, a permanent repository must be established in this country. Some comments promoted reprocessing as a means to stabilize and prepare the spent nuclear fuel for geologic disposal. ***The Nuclear Waste Policy Act of 1982, as amended, establishes a framework for the ultimate disposition of spent nuclear fuel in the United States in a geologic repository. Any foreign research reactor spent nuclear fuel accepted into the United States under the alternatives considered in the EIS would be eligible for disposal in a geologic repository. Under authority of the Act, DOE is currently evaluating the feasibility of locating a geologic repository at Yucca Mountain in Nevada. In the meantime, however, DOE and the Department of State are seeking to stem the use of highly-enriched uranium in civil programs. Under the preferred alternative, if any foreign research reactor spent nuclear fuel were accepted into the United States, it would be treated and/or packaged, and the resulting materials placed in "road ready" storage pending the availability of a geologic repository, if it were not otherwise disposed of in the meantime.***

S.5.5.3 Transportation and Emergency Response

Transportation and emergency preparedness were key concerns expressed during the public comment period. The majority of comments dealt with identification of parties responsible for responding to an accident involving transport of the foreign research reactor spent nuclear fuel, local emergency response capability, marine and ground transportation routing, shipment methods, procedures, and safety criteria. ***Local and State responders would be the first to respond to a transportation accident involving the foreign research reactor spent nuclear fuel shipments, as they would to any overland shipment involving hazardous materials. State, local, and some Tribal governments have the basic capabilities and training that would be required in order to take initial measures to respond to a transportation accident by virtue of their preparation for responding to accidents involving hazardous***

materials, (i.e., assess the scene, administer emergency care, control the area, and call for a hazardous materials special team). DOE would develop emergency plans with the carrier, port officials, State, local, and Tribal officials and provide training courses for first responders to enhance their capabilities to respond appropriately in the unlikely event of an accident involving these spent nuclear fuel shipments. Technical assistance would also be provided to supplement existing State, local, or Tribal resources if any deficiencies are identified. In the event of an accident, if requested by a State, Tribal, or local government, DOE would send a radiological monitoring assistance team from the closest of eight DOE regional offices located across the country.

Appendix H, which was added to the Final EIS in response to public comments, contains the general provisions for emergency preparedness and security measures associated with the transportation of foreign research reactor spent nuclear fuel in the United States. The provisions include communications and meetings between DOE and State, Tribal, and local authorities, prior to the implementation of the policy, for the identification and resolution of emergency management and security issues specific to the communities that would be affected. These issues include capabilities and training of first emergency responders.

Many commentors were concerned about the safety of transportation casks. ***Spent nuclear fuel is transported in "Type B" transportation casks that are designed and built to preclude release of radioactive material. They are subject to stringent design, fabrication, and operating requirements imposed by the Nuclear Regulatory Commission and Department of Transportation in the United States and by the International Atomic Energy Agency for international shipments, to withstand very severe accidents without releasing their contents. These casks are required to be able to pass stringent tests, including a 30-foot drop onto an unyielding surface (such surfaces are engineered and built for these tests and do not exist in nature), a drop onto a steel post (a puncture test), and a high temperature fire test. As a result of their very robust design and construction, to date, no "Type B" spent nuclear fuel transportation cask has ever been punctured, nor has one ever released its radioactive contents, even as the result of an accident.***

Comments on land transportation dealt mostly with routing concerns and emergency response. Several commentors requested that DOE provide notification to local officials and private citizens of the specific routes that would be used for truck or rail shipments. Many commentors expressed concern regarding the risks associated with the use of specific routes (major interstates through population centers) and during adverse weather and traffic conditions. Some questioned the safety records of radioactive waste trucking firms and inquired about the safety requirements imposed on these firms and the contract arrangements that DOE would make with the shippers. ***As part of the development of a Transportation Plan (in which State, local and Tribal officials in addition to DOE, the carrier, shippers agent, the port and other Federal agencies would be involved), highway routes would be identified using criteria developed by the Department of Transportation. These criteria include using the Interstate highway system, selecting the shortest route or time in travel from the U.S. port of entry to the closest Interstate, and using by-passes or beltways to avoid major population centers. States and Tribes may designate alternate routes that are equivalent to the Interstate system in consultation with local officials, and approved by the U.S. Department of Transportation. Rail routing criteria used by the Department include avoiding interchanges and using the best available track. NRC approval of either rail or truck routes selected for use would be required. Official notification of the shipments would be provided to the Governor of each***

State and Governors or Chairpersons of Indian tribes along the route at least seven days in advance of shipment. In addition, DOE would use a satellite-based tracking system to notify Tribes and States of the pending shipment and to continuously track shipment progress. In order to maintain security, Governors and Tribal leaders are required by the NRC to only notify State and local officials who would need to know about the shipment, usually emergency management or law enforcement officials. With respect to the safety record of potential trucking firms, DOE has developed and implemented a mandatory Motor Carrier Evaluation Program with twelve evaluation criteria. Under the Motor Carrier Evaluation Program criteria, trucking firms with poor safety records would be excluded from transporting the spent nuclear fuel. The Motor Carrier Evaluation Program would be invoked as one of the requirements in DOE's foreign research reactor spent nuclear fuel acceptance contract. Other requirements would be discussed during the development of the Transportation Plan with the appropriate State, local, and Tribal officials.

Many commentors requested coordination with emergency responders en route so that localities can be prepared in the unlikely case of an accident. Many State, Tribal, and local representatives, as well as private citizens, commented that communities along shipping routes and at port and management site locations may have inadequate capabilities to respond to emergencies involving radioactive release. Many expressed the need for DOE funding for training, equipment, monitoring for local emergency responders, transportation plans, and real-time shipment tracking that would be accessible to emergency response personnel. A number of commentors suggested that the Final EIS should evaluate the potential impact on local services due to the financial burden associated with emergency response preparedness. ***DOE is committed to working with State, Tribal, and local governments to ensure that they are prepared to carry out their responsibilities in the unlikely event of an accident involving shipment of foreign research reactor spent nuclear fuel. Details of emergency preparedness, security, and coordination of DOE with local emergency response authorities would be contained in the Transportation Plan, which would be prepared prior to any individual spent nuclear fuel shipment and coordinated with State, Tribal, and local officials. Any additional training or equipment needed would be provided as part of the planning process. In addition to direct Federal assistance to State, Tribal, and local governments for maintaining emergency response programs, there are three national emergency response plans under which DOE provides radiological monitoring and assessment assistance. Under these plans, DOE provides technical advice and assistance to the State, Tribal, and local agencies who might be involved in responding to a radiological incident.***

Another group of commentors expressed concern regarding risks of terrorist activities. Several noted that terrorist activity is a concern of all countries, including the United States, citing the Oklahoma City bombing incident as an example. Commentors also stated that transporting nuclear material overseas to the United States would unnecessarily expose shipments to an increased possibility of terrorist threat. ***In response to these concerns, Section D.5.9 was added to Appendix D of the EIS to specifically address terrorism and sabotage. This section concludes that while the risk of certain terrorist and sabotage attempts cannot be precluded, proper security measures would greatly reduce the risk. All shipments of foreign research reactor spent nuclear fuel would be conducted meeting, or exceeding, all the relevant security requirements in the Code of Federal Regulations. DOE would ensure through the spent nuclear fuel acceptance contracts with the reactor operators that proper security is provided at a port or in transit, based on the Nuclear Regulatory Commission requirements. Often local or State law enforcement personnel would be employed***

by the carrier to satisfy these security requirements, which include having armed escorts on board or near the shipment when it is in highly populated areas or at the port in the United States. In the case of military ports, a high level of security is inherently in place.

With regard to marine transport, many commentors stated a preference for using special purpose, chartered, or military ships rather than regularly-scheduled commercial liners to ship spent nuclear fuel. ***The use of commercial liners, chartered ships, and purpose-built ships was considered for the marine transport of the spent nuclear fuel. The analyses in the EIS indicate that the impacts associated with the use of any of the ships evaluated would be small. The impacts of using military ships were not analyzed in the EIS because DOE believes that the added security provided by such ships would not be required to ensure safe transport. DOE's preferred alternative includes the use of military ports as points of entry to the United States.*** Independent inspections by State, local, and/or public interest groups prior to and during shipments were suggested by some commentors. ***DOE would encourage inspections by authorized State agencies for both radiological and vehicle inspections prior to shipment and after arrival at the management site. These inspections would be coordinated with the States through the transportation planning process.***

S.5.5.4 Port Selection Criteria and Activities

Many commentors, predominately those from communities at or near potential ports of entry, questioned DOE's port selection process and the methods for application of the selection criteria, especially with respect to populations in and around candidate ports. Particular concerns were that longshoremen may not be adequately trained to handle radioactive materials or that they could be exposed to high levels of radioactivity. As an alternative, military ports were supported as having the necessary experience in handling nuclear material and being more secure. ***Section 3151 of Public Law 103-160 (the National Defense Authorization Act for the fiscal year 1994), requires that "the Secretary of Energy shall, if economically feasible and to the maximum extent practicable, provide for the receipt of spent nuclear fuel... at a port of entry in the United States which...had the lowest human population in the area surrounding the port of entry...". While this Act was written specifically to apply only to the receipt and storage of spent nuclear fuel at the Savannah River Site, DOE elected to apply this criterion, among others, to the maximum extent practicable, in identifying all suitable ports of entry for potential receipt of foreign research reactor spent nuclear fuel. In application of the population criterion, DOE considered both the population nearest the potential ports of entry analyzed, and the total population along the transportation routes. Analysis of the list of candidate ports against this criterion did not identify any port as a clear choice. Therefore, DOE selected ports that best met all of the criteria discussed in Appendix D to the EIS (e.g., appropriate experience, favorable transit from open ocean, appropriate facilities, access to intermodal transportation and human population). Both commercial and military ports were evaluated. Based on the results of this analysis, DOE believes that foreign research reactor spent nuclear fuel could be received safely via commercial ports, as it has been in the past. Nevertheless, DOE agrees that the use of military ports would provide additional security over that which would be available in a commercial port. Furthermore, although DOE has committed to provide assistance to State and local authorities to ensure that the longshoremen (or other workers) in a commercial port would have any additional training that might be required to allow them to safely handle the spent nuclear fuel, DOE considers that the personnel at a military ordnance facility would be particularly qualified to handle the spent nuclear fuel by***

virtue of their training and experience in performing their military function. Consideration of all these factors led to designation of the Naval Weapons Stations at Charleston and Naval Weapons Station Concord as the preferred ports of entry.

DOE notes that, although the maximum allowable radiation dose rate is 200 mrem per hour, this limit is applicable at the surface of the transportation cask, which would be inside of the container. The maximum radiation dose rate limit to those that would be near the container, such as longshoremen, is 10 mrem per hour at a distance of 2 meters (6.6 ft) from the surface of the container. The actual total dose that a longshoreman would get handling a cask would be quite small due to the fact that a handler would not be present at the surface of the container for long, and the total time near the cask would be quite short. The additional barrier imposed by the standard shipping container would also prevent the longshoreman from being in the near vicinity of the cask. The analysis in the EIS indicates that both the dose and dose rate for the port workers would be low.

Concerns over possible storage of spent nuclear fuel at the port of entry were raised by a number of commentors. ***DOE's goal would be to minimize holding times at the ports and to provide safe transport of the spent nuclear fuel to its destination as quickly as possible. Under normal circumstances, the foreign research reactor spent nuclear fuel would remain at the port for only a few hours (e.g., 2 to 4 hours) and no more than 24 hours. In the very unlikely event that the spent nuclear fuel could not be moved within 24 hours, special provisions to move the fuel to a secure area at the port would be made. Part of the overall plan and agreements with the Department of Defense would include these special provisions.***

Several commentors pointed to recent increases in marine traffic and industrial congestion in the port areas and questioned whether the selection criteria would be affected by these factors. Some cited the need to consider site-specific factors such as hurricanes, severe winds, seismic activity, extreme weather conditions, and sinkholes. ***In general, the number of ship mishaps is not proportional to the amount of ship traffic because port ship traffic is slow, and even when heavy, is normally a small number of ships per hour. Historically, increasing the volume does not significantly increase the probability of an accident. Rather, the number of ship mishaps is associated with navigational hazards and distances from the port to the open ocean or a large bay. In order to further assure safety, the U.S. Coast Guard would establish a moving zone of exclusion, which would keep all vessels away from the ship bringing the spent nuclear fuel into port. Coast Guard escort boats would accompany the ship to port. As for accidents, the potential consequences of a port or land transport accident due to an earthquake are represented and bounded by the potential port and land transportation accidents that are assessed in the EIS. Local hazards, such as earthquakes, volcanoes, and mud slides could be accident initiators; however, they would not increase the consequences of the accident, which were found to be low. Earthquakes were not analyzed separately in the EIS because seismic activity would not result in greater damage to a transportation cask than that analyzed for accidents such as challenges to the transportation cask integrity that could be caused by casks falling from a bridge or down an embankment. These kinds of accidents are within the design standards developed by the NRC and by which cask designs are evaluated. The NRC certifies the designs that contain the appropriate level of safety to protect workers, the public, and the environment from the radioactive material being transported. Analysis of the potential impacts associated with the possible existence of sinkholes along potential rail routes was added to the Final EIS in response to public comment.***

S.5.5.5 Economic Impacts to Candidate Port and Site Communities

Potential economic impacts on affected port and site communities were the subject of many comments. Of particular concern were the socioeconomic impacts to a community in the event of an accidental release of radiation. Examples of potential impacts cited by commentors include disruptions in normal commerce, loss of business, loss of tourism, devaluation of property, and closure of ports and highway routes. Several port authorities were concerned about the potential for declining business due to the perceived stigma associated with handling nuclear waste materials in their ports, while others viewed handling these shipments positively. The costs of emergency response, cleanup, health care, and potential economic losses associated with accidents or releases were key concerns of several State, Tribal, and local officials. ***The risk associated with shipments of foreign research reactor spent nuclear fuel through any of the ports identified would be less than the risk associated with the handling of other hazardous cargoes due to the rigid criteria established for spent nuclear fuel shipping casks. In fact, no adverse impacts have been observed during the 30 years that foreign research reactor spent nuclear fuel was accepted into the United States. Historically, shipping foreign research reactor spent nuclear fuel through ports has not created a stigma or had an adverse economic impact on business, major industries, tourism, or future business development at ports. DOE does not believe that actions such as permanent road closures would be required for the safe and uneventful transportation of foreign research reactor spent nuclear fuel. Costs of emergency response are covered under insurance that is required of hazardous material carriers. If that level of coverage is exceeded, Price-Anderson and other Federal provisions would cover costs.***

S.5.5.6 Health Effects and Environmental Risks

Many commentors raised concerns about health effects and environmental risks that could result from accidents during marine transport, handling operations at ports, ground transportation, and interim management. Of particular concern were the effects of possible radioactive releases into the ocean and rivers, and on highways and railroads; the impacts to fish, wildlife, ecosystems, and drinking water; and the possibility of an increased risk to workers and the public of cancer and genetic defects. ***Human health and safety were primary considerations during the evaluation of environmental effects for the proposed alternatives. Conservative estimates of radiological and nonradiological impacts indicate that risks to the population and workers would be low. The analysis in the EIS indicates that the risks associated with an accident at sea or a port accident would be low. The impacts of the incident-free receipt, handling, and transportation of foreign research reactor spent nuclear fuel would also be extremely low. In over 40 years of spent nuclear fuel transportation, no "Type B" spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive material contents. DOE believes that spent nuclear fuel transportation casks passing through any of the potential ports of entry or any other part of the country would be highly unlikely (i.e., less than a 1 in 10 million chance) to release their contents or adversely affect air or water quality.***

Several commentors questioned the results of the risk analyses in the EIS, suggesting that DOE may have underestimated the risk potential for accidents, radioactive release(s), and exposures to both workers and the public. ***DOE believes that the analyses of risk to people during marine transport and for those who live near potential ports of entry, along transportation routes, and near management sites are conservative (i.e., are likely to overstate the actual risk). These estimates were generated using standard computer codes (e.g.,***

RADTRAN) that have been adopted and used by the Nuclear Regulatory Commission and Department of Transportation for transportation calculations for over 19 years. These computer codes are available for public review.

Some commentors expressed concern about potential health impacts resulting from a cask sinking in deep waters. Many challenged the applicability of the severe accident tests applied to the casks (e.g., crash, fire, drop, immersion), stating that the conditions of real-life accidents were of greater magnitude than the conditions in the tests. For example, commentors cited fires that were alleged to have burned longer and hotter than those used to test the transportation cask and pointed out that the water in Puget Sound is deeper than the cask recovery depth cited in the Draft EIS. ***The EIS presents an evaluation of the consequences of accident scenarios that would result in the sinking of a spent nuclear fuel transportation cask on the continental shelf (water depth of about 200 meters), and in the deep ocean (water depth of more than 200 meters). In the unlikely event that a transportation cask loaded with foreign research reactor spent nuclear fuel were to sink in any U.S. coastal or inland waters, it would be recovered, even from the deepest portions of the Puget Sound, which reach depths of 305 meters (1,000 feet). The sequence of testing scenarios (i.e., cask drop onto unyielding surface, cask drop on a steel post [puncture], and cask fire) is required by the Nuclear Regulatory Commission as part of the certification of "Type B" spent nuclear fuel transportation casks. These tests conservatively represent a wide range of accident conditions that could occur during transport. The test results indicate that if such accidents were to occur, the cask most likely would not fail, and would not lead to a loss of containment. The cask drop and puncture tests evaluate the resulting impact on the most vulnerable orientation of the cask, and on an unyielding surface, which would be unlikely to occur while the cask was being transported in an International Standards Organization (ISO) approved container. In reference to shipboard fires, the duration of a fire is directly related to the amount of combustibles carried on board. The number of severe fires on ships is relatively small. Data available on the last 15 years from Lloyd's of London indicate that of 1,073 ship collisions in port worldwide, only 11 led to fires, and of those, only 5 caused extensive damage, with only 1 actually causing buckling of structures.***

In regard to the impact a ship fire might have on a spent nuclear fuel transportation cask, there are three facts that mitigate the potential damage. First, ship fires tend to move to different areas of the ship as the combustible material is consumed, so the cask would not be exposed for the entire duration of the fire. Second, a ship fire's intensity is normally limited by the amount of oxygen that can reach the interior of a hold. Third, all ships that would be used to transport foreign research reactor spent nuclear fuel have built-in fire suppression equipment, which at a minimum would keep fires well below the extreme temperatures needed to damage the transportation cask. For these reasons, it is almost impossible for spent nuclear fuel inside the transportation cask to reach 900 degrees Kelvin (1,160 degrees F), the melting point of foreign research reactor spent nuclear fuel. The probability of reaching such a temperature is discussed in Attachment D5 to Appendix D of the EIS.

S.5.5.7 Public Involvement Process

A number of comments dealt with DOE's policies and procedures for conducting the public hearings, the duration of the public comment period, the degree to which comments would be considered by DOE in the decisionmaking process, and general distrust of DOE. Several commentors stated that there was insufficient notice and advertising for the public hearings. Many commentors stated the need for additional time to comment on the Draft

EIS. Commentors at the heavily-attended west coast port hearings tended to favor the more traditional, formal public hearing format, and strongly opposed the use of notetakers to summarize hearing issues. In the Tacoma area, commentors urged DOE to hold another hearing to tape record their comments for the record, without allowing for dialogue with DOE representatives. Many State and local officials requested that DOE provide better advance notification to communities that are being considered as candidate ports or management sites so that they have more time to review the Draft EIS. Many individuals stated they had not received the Draft EIS in a timely manner and consequently, had little time to review and comment. Several commentors expressed a desire for increased DOE interaction with local officials and more community participation in DOE's planning and decisionmaking processes.

Notice of the availability of the Draft EIS for public review and comment was published in the Federal Register (60 FR 19899, April 21, 1995). This notice advised concerned parties, including State, Tribal, and local authorities, of the availability of the Draft EIS and the dates and locations of the public hearings on the Draft EIS. In addition, advertisements of the public hearings were placed in local papers prior to their occurrence. The public hearing format adopted by DOE provided an opportunity for interaction between DOE and the public, thus serving to facilitate communication.

In response to public concerns of insufficient time to review the Draft EIS, DOE extended the deadline for submission of written public comments from June 20 to July 20, 1995. DOE considers that this 90-day period was sufficient for public comment. All oral comments presented at each hearing were summarized and have been addressed along with the written comments in Volume 3 of the Final EIS. DOE considers that these actions have provided ample opportunity for the public to comment. Issues raised by the public during the comment period were considered in selection of the preferred alternative for this proposed action. All comments, written and oral, are part of the public record.

S.5.6 Availability of the EIS

Copies of the EIS and the EIS Summary may be obtained by calling DOE's Center for Environmental Management at 1-800-736-3282 (1-800-7-EM DATA). The EIS and EIS Summary may be reviewed at any of the Reading Rooms identified in this Summary.

General questions concerning the NEPA process, under which EISs are prepared, may be addressed to:

Ms. Carol Borgstrom
Office of NEPA Policy and Assistance (EH-42)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
Telephone (202) 586-4600, or leave message at 1-800-472-2756

Written request for clarifications concerning the Foreign Research Reactor Spent Nuclear Fuel program may be sent to:

Mr. Charles Head, Program Manager
Office of Spent Nuclear Fuel Management
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

S.5.7 Record of Decision

The Record of Decision, to be issued no less than 30 days after the Environmental Protection Agency publishes a *Federal Register* Notice of Availability for the Final EIS, will document the decisions made by DOE and the Department of State after the evaluation of the potential environmental impacts of the range of alternatives and appropriate non-environmental factors.

S.5.8 DOE Reading Rooms

A complete copy of the Final EIS and a list of reference documents may be reviewed at any of the public Reading Rooms and information locations listed below.

- Department of Energy Reading Rooms -

Public Reading Room for U.S. Department of Energy Headquarters

Room 1E-190, Forrestal Building
Freedom of Information Reading Room
1000 Independence Avenue, SW
Washington, DC 20585

(202) 586-6020

Public Reading Room for U.S. Department of Energy Oakland Operations Office

Environmental Information Center
1301 Clay Street, Room 700 N
Oakland, CA 94612

(510) 637-1762

Public Reading Room for U.S. Department of Energy Rocky Flats Operations Office

Front Range Community College Library
3645 W. 112th Avenue, Level B
Westminister, CO 80030

(303) 469-4435

Public Reading Room for U.S. Department of Energy Idaho Operations Office

Public Reading Room
1776 Science Center Drive
Idaho Falls, ID 83402

(208) 526-9162

Public Reading Room for U.S. Department of Energy University of Illinois at Chicago Library

Government Documents Section
801 South Morgan Street
Chicago, IL 60607

(312) 996-2738

Public Reading Room for U.S. Department of Energy National Atomic Museum

87117 Wyoming Boulevard, SE (Kirtland AFB)
Albuquerque, NM 87185

(505) 845-4378

Public Reading Room for U.S. Department of Energy Nevada Operations Office

Coordination and Information Center
3084 South Highland Drive
P.O. Box 98521
Las Vegas, NV 89106

(702) 295-0731

Public Reading Room for U.S. Department of Energy Fernald Operations Office

Public Environmental Center
JANTER Building 10845
Hamilton-Cleves Highway
Harrison, OH 44503

(513) 738-0164

Public Reading Room for U.S. Department of Energy Savannah River Operations Office

DOE Public Reading Room
University of South Carolina - Aiken Campus
Grigg-Graniteville Library
2nd Floor
171 University Parkway
Aiken, SC 29801

(803) 641-3320

Public Reading Room for U.S. Department of Energy Oak Ridge Operations Office

Public Reading Room
55 Jefferson Avenue
Oak Ridge, TN 37831

(615) 576-1216

Public Reading Room for U.S. Department of Energy Richland Operations Office

Washington State University Tri-Cities
100 Sprout Road, Room 130 West
Richland, WA 99352

(509) 376-8583

- Other Locations -

Concord Branch Library
2900 Salvio Street
Concord, CA 94519
(510) 646-5455

George A. Smathers Libraries, Library West
University of Florida Library, Room 241
P.O. Box 11701
Gainesville, FL 32611-7001
(904) 392-0367

Jacksonville Public Library
Documents Department
122 North Ocean Street
Jacksonville, FL 32202
(904) 630-2665

Atlanta Public Library
Government Documents Section
1 Margaret Mitchell Square
Atlanta, GA 30303
(404) 730-1700

Reese Library
Augusta College
2500 Walton Way
Augusta, GA 30904-2200
(706) 737-1744

Chatham-Effingham-Liberty Regional Library
2002 Bull Street
Savannah, GA 31401
(912) 234-5127

Boise Public Library
Government Documents Section
715 South Capitol Boulevard
Boise, ID 83702
(208) 384-4023

INEL Oversight Program Library
Idaho Department of Health & Welfare
1410 North Hilton, Third Floor
Boise, ID 83706
(208) 334-0498

Idaho Falls Public Library
457 Broadway
Idaho Falls, ID 83402
(208) 529-1462

Pocatello Public Library
812 East Clark Street
Pocatello, ID 83201
(208) 232-1263

Twin Falls Public Library
Reference Desk
434 Second Street East
Twin Falls, ID 83301
(208) 733-2964

Amargosa Valley Community Library
HCRoute 69, Box 401-T
829 Farm Road
Amargosa Valley, NV 89020
(702) 372-5340

Carson City Public Library
900 North Roop Street
Carson City, NV 89701
(702) 887-2244 or (702) 887-2245

Nye County Nuclear Waste Repository Project Office
P.O. Box 1767
475 St. Patrick Street
Tonopah, NV 89049
(702) 482-8183

Brunswick County Government Center
Mr. Wyman Yelton, City Manager
P.O. Box 249
45 Court House Drive, NE
Bolivia, NC 28422
(910) 253-4331

Pembroke State University Library
1 University Drive
Pembroke, NC 28372
(910) 521-6265

- Other Locations (Continued) -

D. H. Hill Library

North Carolina State University
P.O. Box 7111
Raleigh, NC 27695-7111

(919) 515-3364

New Hanover County Public Library

Attn: Daniel Horn
201 Chestnut Street
Wilmington, NC 28401

(910) 341-4390

Brantford Price Millar Library

Portland State University
934 S.W. Harrison
Portland, OR 97201

(503) 725-4617

Charleston County Main Library

404 King Street
Charleston, SC 29403

(803) 723-1645

South Carolina State Library

1500 Senate Street
Columbia, SC 29201

(803) 734-8666

Berkeley County Library

100 Library Street
Monks Corner, SC 29461

(803) 722-3550

Otranto Regional Library

2261 Otranto Road
North Charleston, SC 29418

(803) 572-4094

Clinton Public Library

118 South Hicks Street
Clinton, TN 37716

(615) 457-0519

Lawson McGhee Public Library

500 West Church Avenue
Knoxville, TN 37902

(615) 544-5750

Memphis/Shelby County Public Library and Information Center

1850 Peabody Avenue
Memphis, TN 38104

(901) 725-8800

Oak Ridge Public Library

Civic Center
Oak Ridge, TN 37830

(615) 482-8455

Rosenberg Library

Attn: Judy Young
2310 Sealy Avenue
Galveston, TX 77550-2296

(409) 763-2526

Houston Public Library

Attn: Social Sciences
500 McKinney
Houston, TX 77002

(713) 247-2222

Hampton Public Library

4207 Victoria Boulevard
Hampton, VA 23669

(804) 727-1154

Newport News Public Library

Grissom Branch
366 Deshazor Drive
Newport News, VA 23602

(804) 886-7896

Kirn Library

301 East City Hall Avenue
Norfolk, VA 23510

(804) 441-2429

- Other Locations (Continued) -

Portsmouth Public Library

Main Branch
601 Court Street
Portsmouth, VA 23704

(804) 393-8501

Owen Science & Engineering Library

Washington State University
Pullman, WA 99164-3200

(509) 335-4181

Seattle Public Library

1000 Fourth Avenue
Seattle, WA 96104

(206) 386-4636

Suzallo Library, SM25

University of Washington Libraries
University of Washington
Seattle, WA 98185

(206) 543-9158

Foley Center

Gonzaga University
East 502 Boone Avenue
Spokane, WA 99258

(509) 328-4220, Extension 3125

Pierce County Library

300 512th Street, East
Tacoma, WA 98446

(206) 536-6500

Tacoma Public Library

1102 Tacoma Avenue South
Tacoma, WA 98402

(206) 591-5666

DOE/EIS-0218F
February 1996
Volume 1

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Department of Energy
Washington, DC 20585

February 8, 1996

Dear Interested Party:

I am enclosing a copy of the final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel. The Department of Energy, in cooperation with the State Department, prepared the final Environmental Impact Statement.

This study analyzes the potential environmental impacts of adopting a policy to manage foreign research reactor spent fuel containing uranium enriched in the United States. In particular, the study examines the comparative impacts of several alternative approaches to managing the spent fuel. The analyses demonstrate that the impacts on the environment, workers and the general public of implementing any of the alternative management approaches would be small and within applicable Federal and state regulatory limits.

The Department's preferred approach to managing the spent fuel, referred to in the study as the "preferred alternative," is for the Department to receive the spent fuel into the United States, and to manage it at the Department's Savannah River Site in South Carolina and the Idaho National Engineering Laboratory. The spent fuel would be shipped to the United States over 13 years through two military ports. The Charleston Naval Weapons Station in South Carolina would receive about one to two shipments every month beginning in 1996. The Concord Naval Weapons Station in California would receive far fewer shipments (as few as five shipments over a 13-year period) beginning in 1997.

The final Environmental Impact Statement is a three-volume document, approximately 4000 pages in length. Volume 1 (494 pages) describes the policy considerations of adopting a policy to manage foreign research reactor spent fuel, and the potential environmental impacts. Volume 2 (1111 pages) contains eight appendices relating to the technical analyses. Volume 3 (2230 pages) contains the public's comments on the draft Environmental Impact Statement, the Department's responses to those comments, and summaries of the 17 public hearings held throughout the United States during the 90-day comment period on the draft.

If you would like another copy of the entire study, a particular volume, or an additional copy of the Summary, we would be pleased to send it to you. Please let us know by calling the Department's Center for Environmental Management Information at 1-800-736-3282 (toll-free). The entire document will be placed in the public reading rooms and information locations listed in the Summary.

The Department will not make a final decision on whether to adopt the proposed policy until late March 1996. Thank you for your interest in this proposed action.

Sincerely,

Thomas P. Grumbly

Thomas P. Grumbly
Assistant Secretary for
Environmental Management

Enclosure

DOE/EIS-0218F
February 1996

Volume 1

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Cover Sheet

Responsible Agencies: Lead Agency: United States Department of Energy
Cooperating Agency: United States Department of State

Title: Final Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Contact: For further information, concerning this Final Environmental Impact Statement, contact:

Charles Head, Program Manager
Office of Spent Nuclear Fuel Management (EM-67)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585

For general information on the United States Department of Energy's National Environmental Policy Act process, call 1-800-472-2756 to leave a message, or contact:

Carol Borgstrom, Director
Office of NEPA Policy and Assistance (EH-42)
U.S. Department of Energy
1000 Independence Avenue, SW
Washington, DC 20585
202-586-4600

Abstract: The United States Department of Energy and United States Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, by seeking to reduce and eventually eliminate highly-enriched (weapons-grade) uranium from civilian commerce worldwide. Environmental effects and policy considerations of three Management Alternative approaches for implementation of the proposed policy are assessed. The three Management Alternatives analyzed are: (1) acceptance and management of the spent nuclear fuel by the Department of Energy in the United States, (2) facilitate the management of the spent nuclear fuel at one or more foreign facilities (under conditions that satisfy United States nuclear weapons nonproliferation policy objectives), and (3) a combination of elements from one or both of Management Alternatives 1 and 2 (Hybrid Alternative). A No Action Alternative is also analyzed.

For each Management Alternative, there are a number of implementation alternatives. For Management Alternative 1, this document addresses the environmental effects of various implementation alternatives, such as varied policy durations, management of various quantities of spent nuclear fuel, chemical separation, developmental treatment and/or packaging technologies, and differing financing arrangements. Environmental impacts are also examined at various potential ports of entry, along truck and rail transportation routes, at candidate management sites, and for alternate storage technologies. For Management Alternative 2, this document addresses the environmental effects of two implementation alternatives: (1) assisting foreign nations with storage; and (2) assisting foreign nations with reprocessing

of the spent nuclear fuel. With respect to Management Alternative 3, an example Hybrid Alternative is analyzed wherein a portion of the spent nuclear fuel would be processed at overseas facilities and the remaining portion would be managed in the United States.

The United States Department of Energy and United States Department of State, in consultation with other government agencies, designate the acceptance and management of the foreign research reactor spent nuclear fuel in the United States (i.e., Management Alternative 1 with modifications to several basic implementation elements) as the preferred alternative.

Public Comments: The public comment period on the Draft EIS was conducted from April 21, 1995 to July 20, 1995. During this period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and 5 candidate spent nuclear fuel management sites. In addition, a public hearing was held in Washington, D.C. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

Foreword

This Final Environmental Impact Statement presents an evaluation of policy considerations and potential environmental impacts resulting from the U.S. Department of Energy and the U.S. Department of State joint proposal to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel that contains uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy would be to promote nuclear weapons nonproliferation objectives of the United States, specifically by seeking to reduce, and eventually to eliminate, highly-enriched (weapons-grade) uranium from civil commerce worldwide. This policy is jointly proposed by the U.S. Department of Energy and the U.S. Department of State. This document was prepared in compliance with the National Environmental Policy Act and in accordance with regulations issued and published by the Council on Environmental Quality (40 CFR Parts 1500-1508) and the U.S. Department of Energy (10 CFR Part 1021).

Environmental effects and policy considerations of several alternative approaches for implementation of the proposed policy are assessed. Three Management Alternatives are analyzed: (1) acceptance and management of the spent nuclear fuel by the Department of Energy in the United States; (2) facilitate the management of the spent nuclear fuel at one or more foreign facilities under conditions that satisfy United States nuclear weapons nonproliferation policy objectives; and (3) a combination of components of Management Alternatives 1 and 2 (Hybrid Alternative Example). A No Action Alternative is also analyzed.

For each Management Alternative, there are a number of alternatives for its implementation. For Management Alternative 1, this document addresses the policy implications and environmental effects of various implementation alternatives such as varied policy durations, management of various quantities of spent nuclear fuel, and differing financing arrangements. Environmental impacts at various potential ports of entry, along truck and rail transportation routes, at candidate management sites, and for alternate storage technologies are also examined. For Management Alternative 2, this document addresses two subalternatives: (1) assisting foreign nations with storage; and (2) assisting foreign nations with reprocessing of the spent nuclear fuel. With respect to Management Alternative 3, a hybrid alternative example is analyzed, utilizing the analysis provided for Management Alternatives 1 and 2, wherein a portion of the spent nuclear fuel would be processed at overseas facilities and the remaining portion would be managed in the United States.

A Notice of Intent to prepare this document was published in the Federal Register on October 21, 1993. Nine public scoping meetings were conducted during November and December of 1993. The period for acceptance of public comments on this document closed on December 8, 1993. However, the United States Department of Energy continued to accept written comments through January 31, 1994. In October 1994, the Implementation Plan for this Environmental Impact Statement was issued to provide guidance for its preparation and to record the U.S. Department of Energy's disposition of comments received during the scoping process.

The Draft Environmental Impact Statement was issued in April 1995. The public comment period on the Draft Environmental Impact Statement was from April 21, 1995 to July 20, 1995. During this period, DOE held 17 public hearings in the locations most likely to be directly affected by the EIS alternatives, including the 10 candidate ports of entry and 5 candidate spent nuclear fuel management sites. In addition, a public hearing was also held in Washington, D.C. The Draft EIS was made available to the public through mailings, requests to DOE's Environmental Management Information Center, and at DOE Public Reading Rooms and other designated information locations.

Results of the environmental analyses are presented in two volumes. Volume 1 is composed of eight chapters. Chapter 1 gives the background description of the United States nuclear weapons nonproliferation policy and describes the purpose and need for the proposed action. Chapter 2 then states the proposed policy and describes the three Management Alternatives for its implementation. It includes a discussion of the basic implementation components of Management Alternative 1, as well as implementation alternatives that vary one component of the basic implementation of Management Alternative 1. The implementation alternatives include variations on the duration of the policy, alternative amounts of material that might be covered by the policy, and various financing alternatives. The potential ports of entry, transportation routes, candidate spent nuclear fuel management sites and storage technologies are also described. This chapter also describes Management Alternative 2, which contains two subalternatives for its implementation. Subalternative 1 is to provide assistance to foreign nations with storage of the spent nuclear fuel. Subalternative 2 is to provide assistance with reprocessing of the spent nuclear fuel at one or more foreign locations. Management Alternative 3 is also discussed in this Chapter by tiering off the evaluation and analyses provided for Management Alternatives 1 and 2. The potentially affected environment under Management Alternatives 1 and 3 is described in Chapter 3. Essential results of the environmental analyses are then given in Chapter 4, which summarizes the methods used in the evaluation and provides an assessment of the environmental effects. Details of the environmental analyses are provided in the appendices, which comprise Volume 2 of this document. Chapter 5 describes applicable laws, regulations, and other requirements. A list of the preparers of this Final Environmental Impact Statement, agencies consulted, and references are provided in Chapters 6, 7, and 8, respectively. In addition to these two volumes, a Volume 3 (Comment Response Document) has been added to the Final Environmental Impact Statement which contains the written and oral comments received during the public comment period for the Draft Environmental Impact Statement.

In consideration of public comments, DOE has added information to the EIS including: clarification of the proposed U.S. policy on accepting spent nuclear fuel from allies; examination of the consequences of sabotage or terrorist attack; safety of transportation casks; re-examination of the shipboard fire analysis, and general provisions of transportation and emergency response regulations and management. The Naval Weapons Station at Charleston was analyzed in addition to the other terminals of the Port of Charleston within the greater Charleston area that were discussed in the Draft Environmental Impact Statement.

This Final Environmental Impact Statement has a two-fold purpose. The first purpose is to provide decision makers in the U.S. Department of Energy and the U.S. Department of State with an evaluation of the environmental effects of these policies. The second purpose is to inform the public concerning the essential features, policy considerations, and potential environmental effects of the proposed policy, and to provide the public an opportunity to provide feedback to the U.S. Department of Energy and the U.S. Department of State on the proposed policy.

Reader's Guide

In response to comments submitted after issuance of the Draft Environmental Impact Statement in April 1995, and due to additional technical and policy details not available at the time of issuance of the Draft Environmental Impact Statement, Volumes 1 and 2 of the Final Environmental Impact Statement contain revisions and changes. The revisions and changes made since issuance of the Draft Environmental Impact Statement are indicated by a line in the margin of Volumes 1 and 2. A new Appendix H has been added to Volume 2 to describe the general provisions associated with transportation planning for potential shipments of foreign research reactor spent nuclear fuel. In addition, Volume 1 and each appendix in Volume 2 provide a unique reference list to enable the reader to further review and research selected topics. The U.S. Department of Energy has established reading rooms and information locations across the United States where these references may be reviewed or obtained for review through interlibrary loan. The addresses and phone numbers for these reading rooms and information locations are provided at the end of the accompanying Summary.

Table of Contents

Volume 3

List of Figures

List of Tables

Acronyms and Abbreviations

BNFP	Barnwell Nuclear Fuels Plant
CFR	Code of Federal Regulations
Ci	Curie
cm	centimeter
DOE	Department of Energy
EDE	Effective Dose Equivalent
EIS	Environmental Impact Statement
E-MAD	Engine Maintenance and Disassembly
FAST	Fluorinel Dissolution and Fuel Storage
FMEF	Fuel Maintenance and Examination Facility
g	gram
ha	hectare
HEU	Highly-Enriched Uranium
ICPP	Idaho Chemical Processing Plant
IFSF	Irradiated Fuel Storage Facility
ISO	International Organization for Standardization
kgTM	kilograms of Total Mass
km	kilometer
l	liter
LCF	latent cancer fatality
LEU	Low Enriched Uranium
m	meters
MACCS	MELCOR Accident Consequences Code System
MEI	Maximally Exposed Individual
mg	milligram
mg/l	milligrams per liter
mi	mile
min	minute
ml	milliliter
mm	millimeter
MOTSU	Military Ocean Terminal at Sunny Point
mrem	millirem
MTHM	Metric Tons of Heavy Metal
MTR	Material Test Reactor
NEPA	National Environmental Policy Act
NPAI	Nearest Public Access Individual
NRC	Nuclear Regulatory Commission
NWS	Naval Weapons Station
ppt	parts per thousand
rad	radiation absorbed dose
RBOF	Receiving Basin for Offsite Fuels
rem	roentgen equivalent man
RERTR	Reduced Enrichment for Research and Test Reactors
SNF&INEL Final EIS	Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement
TRIGA	Training, Research, Isotope, General Atomic reactors

1. Introduction

Reducing the threat of the proliferation of nuclear weapons is one of the foremost goals of the United States. Proper management of spent nuclear fuel from foreign research reactors is essential to the efforts aimed at achieving these goals since much of this fuel contains highly-enriched uranium[1] (HEU), which can be directly used in simple nuclear weapons.

The concern over appropriate management of foreign research reactor spent nuclear fuel was reiterated in the Presidential Directive on Nonproliferation and Export Controls, issued by President Clinton on September 27, 1993. In particular, the Presidential Directive included the following language: "We will also seek to minimize the use of highly-enriched uranium in civil nuclear programs. To this end, the Secretary of Energy will review the need for programs to develop alternative fuels for research reactors and accelerate steps towards implementation of a policy of taking back U.S.-origin spent fuels from foreign research reactors."

1.1 Policy Background

Since 1945, every U.S. Administration has recognized that preventing the further spread of nuclear weapons must be a fundamental national security and foreign policy objective of the United States. The initial U.S. approach to nuclear technology was to classify all nuclear activities. However, the United States soon realized that it would be impossible to prevent other nations from acquiring nuclear technology.

Consequently, since the 1950's, beginning with the "Atoms for Peace" program, the United States has provided peaceful nuclear technology to foreign nations in exchange for their promise to forego development of nuclear weapons. In addition, the United States requires that any nuclear technology provided shall be subject to international safeguards and inspections to prevent diversion of materials or technology to nuclear weapons activities.

The Atomic Energy Act of 1946 was the first U.S. legislation regulating nuclear activities. The Act prohibited international nuclear cooperation until effective international safeguards were in place to prevent such cooperation from assisting in the development of foreign nuclear weapons programs. A major revision of the Atomic Energy Act in 1954 provided that foreign countries receiving nuclear assistance had to accept conditions on its use, including making a pledge not to use nuclear materials or equipment provided by the United States for military purposes.

Simply put, peaceful nuclear cooperation, under international safeguards, has been a critical component of U.S. nuclear weapons nonproliferation policy since the beginning of the atomic age. The intent of such peaceful nuclear cooperation is to prevent the development of nuclear weapons programs worldwide.

A major element of the "Atoms for Peace" program for peaceful nuclear cooperation, particularly in the early years, was the provision of research reactor technology and the HEU necessary to fuel the research reactors. Research reactors play a vital role in important medical, agricultural, and industrial applications,

[1] *Uranium enriched to 20 percent or greater in isotope 235 is known as HEU.*

and also provide a tool for fundamental scientific research. For example, research reactors are a vital tool in cancer therapy and radioimmunoassay blood testing. There are approximately 30,000 medical procedures per day in North America, 8,000 to 10,000 procedures per day in Europe, and 8,000 to 10,000 procedures per day on other continents using medical isotopes produced in research reactors in other countries. Neutron radiography provided by research reactors has enabled researchers to diagnose defects in metals and engines of many varieties and to conduct research on new materials, computer chips, and chemicals. Radioisotopes produced in research reactors have been used in leak detection in industrial components and equipment, aluminum production, and semiconductor and solar panel research. Neutron scattering experiments done in research reactors have provided insight into the biostructure of organic substances and have advanced the development of magnetic and superconducting materials. Research reactors have also been used in the environmental sciences to study waste migration, mine drainage, diffusion and transport of pollutants, water chemistry, sediment transport, atmospheric dispersion, and toxic waste management. Another important use of research reactors is irradiation testing of materials and fuel forms, including safety experimentation, to support advanced fuel design and waste management development for use in the power industry. Research reactors also have served as major training facilities in nuclear technology. For example, the research reactor operating in Austria is used by the International Atomic Energy Agency to train personnel who conduct international inspections of weapons and civil nuclear facilities worldwide.

The transfer of enriched uranium from the United States to other nations under the "Atoms for Peace" program was usually supported by a bilateral research agreement for each foreign research reactor. Before 1964, these agreements provided for the lease of the enriched uranium, with explicit provision for the return of the spent nuclear fuel to the United States. After 1964, most agreements provided for the sale of this material to the foreign nation.

After its use (irradiation) in a research reactor, the used (spent) fuel was generally returned to the United States where it was reprocessed to extract the uranium still remaining in the spent fuel. In this way, the United States maintained control over the HEU, which otherwise could be used in the production of nuclear weapons. The United States began accepting HEU spent nuclear fuel from foreign research reactors in 1958.

After 1964, the operative policy under which the United States accepted foreign research reactor spent nuclear fuel containing uranium enriched in the United States became known as the "Off-Site Fuels Policy." This policy was implemented through a series of *Federal Register* Notices issued until 1987, and was incorporated into bilateral international agreements with recipient countries. The term "Off-Site Fuels Policy" was used to indicate that the spent nuclear fuel had been irradiated at facilities not owned by the Department of Energy (DOE). Under the "Off-Site Fuels Policy," the United States accepted, temporarily stored, and reprocessed spent nuclear fuel containing HEU enriched in the United States. The rationale for the policy was to discourage the stockpiling abroad of spent nuclear fuel containing HEU and to recover the fuel value of the HEU remaining in the spent nuclear fuel.

In response to increasing congressional and public concern about the potential diversion of HEU for use in nuclear weapons by foreign nations, subnational groups, or terrorist organizations, DOE in 1978 initiated the Reduced Enrichment for Research and Test Reactors (RERTR) program. The RERTR program was aimed at reducing the use of HEU in civilian programs by promoting the conversion of foreign research reactors from HEU fuel to low enriched uranium (LEU) fuel. Research reactors are of particular interest in this endeavor because the major civilian use of HEU is as fuel in nuclear research reactors.

As a part of the RERTR program, DOE developed LEU fuel and worked with foreign research reactor operators to modify their reactors to run on such fuel. The foreign research reactor operators who converted to LEU fuel did so in support of nuclear weapons nonproliferation objectives, even though such conversions were expensive and generally resulted in reductions in the capabilities of the reactors and increased operating costs.

From the beginning of the RERTR program, foreign research reactor operators made it clear that their willingness to convert their research reactors to LEU fuel was contingent upon the continued acceptance by DOE of their spent nuclear fuel for disposition in the United States. In 1986, to further encourage foreign research reactor operators to convert to LEU fuel, the DOE "Off-Site Fuels Policy" was extended to include the acceptance of spent nuclear fuel containing LEU enriched in the United States.

The RERTR program has been highly successful and many foreign research reactors have been modified to operate, or have been designed to operate, with the high-density LEU fuels developed by the RERTR program, instead of HEU fuel. Of the 42 foreign research reactors with power levels equal to or above one million watts that use U.S. enriched fuel, 37 could operate with the currently available high-density LEU fuels. Of these, 25 are either operating on LEU fuel, or have ordered LEU fuel, and DOE anticipates that an additional eight reactors will convert to LEU fuel by 2001. Work is underway to develop improved high-density LEU fuels that would enable the remaining HEU fueled reactors to convert as well. Thus, the RERTR program has contributed to a significant reduction in the use of HEU in foreign research reactors.

The RERTR program is also developing the technology necessary to substitute LEU for the HEU in targets that are currently irradiated in reactors to produce the radioisotope molybdenum-99 for use as a diagnostic tool in nuclear medicine. The current limited nuclear commerce in HEU for medical targets can be reduced and eventually eliminated when LEU targets become available. When combined, these RERTR program activities can virtually eliminate the need for civilian commerce in HEU.

In 1988, DOE's "Off-Site Fuels Policy" to accept HEU spent nuclear fuel expired. At the end of 1992, the policy as it applied to the acceptance of LEU spent nuclear fuel also expired. The "Off-Site Fuels Policy" was not immediately renewed because of the need to assess the environmental impacts of a new policy. Because the United States has not been in a position to accept HEU fuel for 6 years [except for two recent "urgent relief" shipments of 252 spent nuclear fuel elements, 153 of which were from Denmark, Austria, Sweden, and the The Netherlands, with the remaining 99 elements from Switzerland and Greece, conducted under DOE's "Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel" (DOE, 1994m)], many foreign research reactor operators will soon run out of storage capacity or face safety and regulatory issues associated with the presence of spent nuclear fuel at their sites.

Although those foreign research reactors who could obtain regulatory clearance to build new storage capacity could do so within the duration of the proposed policy, they do not have time to do so in the near term before they run out of space. Storage of the spent nuclear fuel is a concern because it contains both enriched uranium (some of it highly-enriched material suitable for use in nuclear weapons) and highly radioactive waste products. The storage of such spent nuclear fuel must be accomplished with considerable care to ensure that the spent nuclear fuel does not corrode. If it does corrode, it could release fission products within the storage facility (making action to protect the spent nuclear fuel from further degradation difficult). In the extreme, uranium could be released from the spent nuclear fuel and settle to the bottom of the storage facility, creating the potential for a chain reaction.

Even if the spent nuclear fuel is kept in pristine condition (a relatively straightforward task given the resources and determination to store it correctly), the accumulation of large quantities of spent nuclear fuel containing HEU raises the possibility that some of the spent nuclear fuel might be stolen and its uranium diverted into a nuclear weapons program. In addition, spent nuclear fuel storage is an expensive undertaking (partially so because of the steps needed to avoid the problems outlined above) and is limited by local regulation in many countries. As a result, the cessation of the U.S. acceptance of foreign research reactor spent nuclear fuel associated with the expiration of the "Off-Site Fuels Policy" has created significant problems for the research reactor operators and has undercut the perceived reliability of the United States as a partner in peaceful nuclear cooperation, a cornerstone of U.S. nuclear weapons nonproliferation commitments enshrined in the *Treaty on the Non-Proliferation of Nuclear Weapons.*

With respect to the broader role that the United States plays in worldwide peaceful nuclear cooperation, President Reagan warned as early as July 1981, that "if we are not such a partner, other countries will tend to go their own ways and our influence will diminish. This would reduce our effectiveness in gaining the support we need to deal with proliferation problems." (Statement on the U.S. Nuclear Nonproliferation Policy, July 16, 1981.) More recent correspondence from the United States National Security Council, Department of State, Department of Defense, and the Nuclear Regulatory Commission (NRC) underscores the importance and urgency of support for the RERTR program objectives and the need for immediate action to reduce the use of HEU in civil programs. For example, in a recent letter to Secretary of Energy Hazel R. O'Leary, Secretary of State Warren Christopher stated, "The spent fuel acceptance policy which the EIS supports is central to our goal of preventing the spread of nuclear weapons -- and therefore to a major national security objective of this administration" (see Appendix G).

If the United States does not accept the foreign research reactor spent nuclear fuel, some of the foreign research reactors may be forced to shut down, as they will have no way to store any additional spent nuclear fuel. Other research reactor operators may have the option of reprocessing[2] their spent nuclear fuel (separating the uranium from the fission products for use as new fuel) at existing facilities. British and/or French reprocessing plants might accept foreign research reactor spent nuclear fuel for reprocessing, but have done so in the past only on the condition that the reprocessing customer agrees to take back the reprocessing wastes. Some of the countries in which the foreign research reactors are located do not have a domestic waste repository or other facilities for storing the highly-radioactive wastes generated from reprocessing for the near term. Other countries will not allow reprocessing because they object to reprocessing on environmental or nuclear weapons nonproliferation grounds.

While the U.S. Government has full confidence in the physical protection and safeguards systems in place at the British and French reprocessing facilities, reprocessing of spent nuclear fuel containing HEU, as it has been done in the past, would sustain international commerce in HEU, in direct contradiction to the U.S. position on nuclear weapons nonproliferation. It would likely mean that the research reactors pursuing this option would continue operations on the HEU fuel cycle because currently they have a method of disposing of HEU spent nuclear fuel, but not LEU spent nuclear fuel. Neither Dounreay (the British reprocessing facility in Scotland), Marcoule in France, nor any other available facility is currently accepting or has the special equipment that would be used in the United States to reprocess the high density LEU fuels that the United States is encouraging foreign research reactors to use to replace the

[2] *Reprocessing refers to the disassembly of spent nuclear fuel (usually by dissolving it in acid) to allow the uranium, and possibly other fissile materials, to be separated from the fission products and structural parts of the spent nuclear fuel. The fissile materials can then be reused, and the fission products are discharged as waste. These wastes are dissolved in specially formulated molten glass and cast into stainless steel cylinders (or in some cases, in foreign countries, may be mixed with special concretes and poured into steel drums) prior to disposal.*

HEU fuels. Hence, if the research reactors decide to use reprocessing as it has been managed up to this point to prevent backlogs of spent nuclear fuel from building up, they would have to continue to use HEU fuels. This could result in reactor operators delaying or canceling plans to convert to LEU, or in some cases, withdrawing from the RERTR program and reconverting from LEU to HEU fuels. The United Kingdom Atomic Energy Authority is considering adding an LEU processing capability to its plant in Dounreay. At this time, it is not clear that this plant will continue to operate. If the Dounreay facility continues to operate and if an LEU reprocessing capability is installed, that would mean that Dounreay customers could operate their research reactors on LEU.

If some foreign research reactor operators were to withdraw from the RERTR program and rely instead on HEU fuels, with attendant lower costs and enhanced performance, other research reactor operators would be under pressure to convert to the use of HEU for competitive reasons. Since the United States, under the Energy Policy Act of 1992, is barred from exporting HEU to virtually all foreign research reactors, research reactor operators seeking continued use of HEU would be forced to seek alternate suppliers. Russia and China are sources of HEU; and, should they choose to provide a ready supply of HEU, many foreign research reactor operators would be forced to consider abandoning the RERTR program and reconverting to HEU. In addition, the United States is currently attempting to convince Russia and China to implement programs similar to the United States' RERTR program to encourage their nuclear fuel customers to phase out use of HEU in their research reactors. However, if the United States cannot convince those countries to which it has exported nuclear fuel to stop using HEU, the United States would stand little chance of convincing Russia and China to do so with the countries to which they export nuclear fuel.

Additionally, several developed countries involved in the RERTR program are exporters of research reactors. In recent years, they have required that reactors exported to other countries be fueled with LEU. However, if foreign research reactor operators begin delaying or canceling plans to convert to LEU, and thereby continue to use HEU, foreign research reactor purchasers would demand HEU-fueled reactors. This could lead to renewed international commerce in weapons-usable HEU, and would be antithetical to the policy goal of seeking to minimize and eventually eliminate the civil use of HEU.

Another crucial consideration in proposing to accept spent nuclear fuel shipments from foreign research reactors is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The Non-Proliferation Treaty is the basis for the world's nuclear weapons nonproliferation regime. The purpose of the Non-Proliferation Treaty is to keep the number of countries with nuclear weapons from growing. Five countries acknowledge having nuclear weapons: the United States, Russia, the United Kingdom, France, and China. In addition to the five acknowledged nuclear weapons States, 175 other nonnuclear weapons States are members of the treaty. The obligations for compliance with the Non-Proliferation Treaty apply to both nuclear weapons States and nonnuclear weapons States. While nonnuclear weapons States agree not to pursue development or acquisition of nuclear weapons or other nuclear explosive devices, the nuclear weapons States commit to the eventual elimination of their nuclear weapons arsenals and to assist nonnuclear weapons States with peaceful applications of nuclear energy. The Off-Site Fuels Policy and RERTR program are examples of how the United States has helped other nations with peaceful applications of nuclear energy in the past.

The parties to the Non-Proliferation Treaty met in May of 1995 and agreed by consensus to extend the treaty indefinitely and without conditions. Making this vital treaty a permanent part of the international nonproliferation regime was an important U.S. foreign policy achievement. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other

Non-Proliferation Treaty parties that the nuclear weapons States were in compliance with their obligations under Article IV of the Non-Proliferation Treaty to assist the nonnuclear weapons States with applications of nuclear energy for peaceful purposes.

Although the Non-Proliferation Treaty was extended indefinitely at the 1995 Non-Proliferation Treaty Conference, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors, or has been forced to seek reprocessing, could accuse the United States of not having fulfilled its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

In the past, some individuals and groups have incorrectly asserted that the U.S. concerns with reprocessing of HEU spent nuclear fuel discussed above are inconsistent with the U.S. policy of continuing to grant prior consent[3] to Japan and Western European nations for reprocessing of power reactor spent nuclear fuel. The U.S. Government believes that the growing quantities of plutonium in international commerce do present a threat to the efforts of the United States and other countries to prevent the proliferation of nuclear weapons. In countries where material control and accounting or physical protection systems are not sufficiently rigorous, there is a risk of diversion or theft of such materials. In addition, even in countries with effective nuclear weapons nonproliferation commitments, the presence of unneeded stocks of plutonium could raise security concerns on the part of neighboring countries. Accordingly, the U.S. Government does not encourage the civil use of plutonium. Nevertheless, the United States is also committed to being a reliable nuclear trading partner and to avoiding interference in peaceful nuclear programs. Therefore, in Western Europe and Japan where there are well-established civil reprocessing and plutonium facilities and comprehensive nuclear weapons nonproliferation commitments, the United States will continue, in appropriate instances, to grant prior consent for reprocessing of plutonium-bearing spent fuels on a predictable and long-term basis. Undertaking the use of U.S. consent rights to block reprocessing would lead to confrontation with, and would jeopardize the support from, nations that are in accord with the broader U.S. nuclear weapons nonproliferation goals and agenda.

1.2 Purpose and Need For Agency Action

Curbing the spread of nuclear weapons has been an important foreign policy and national security objective of the United States for nearly half a century. The proposed action is one action among many being pursued by the United States to reduce the potential for the proliferation of nuclear weapons. More specifically, the proposed action is intended to support the U.S. policy objective of seeking to reduce, and eventually to eliminate, HEU from civil commerce.

The nuclear weapons proliferation concerns addressed by the proposed action stem from the use of HEU as fuel for foreign research reactors and the presence of large residual amounts of HEU in the spent nuclear fuel from these research reactors. HEU can be used directly in simple nuclear weapons. In the past, the United States has encouraged reductions in the use of HEU as research reactor fuel by conducting

[3] *In general terms, the Atomic Energy Act of 1954, as amended, requires that before the United States may export nuclear material (e.g., enriched uranium), nuclear equipment, or sensitive nuclear technology to another country, that country must agree to obtain the consent (i.e., approval) of the United States before it may transfer that material, equipment, or technology to another country. However, in its 1988 agreement for cooperation with Japan, the United States has granted "prior consent" (i.e., approval in advance) for Japan to transfer power reactor spent fuel to France for reprocessing for the life of the agreement (the initial term is 30 years), subject to certain conditions relating to Japan's nuclear weapons nonproliferation activities. The U.S. Government has made a commitment to include a similar arrangement in new agreements that it is currently negotiating with Euratom and Switzerland.*

the RERTR program (to develop high-density LEU fuels to replace the HEU fuels). The United States also previously prevented the development of foreign stockpiles of HEU in foreign research reactor spent nuclear fuel by conducting the "Off-Site Fuels Policy" (i.e., by accepting the spent nuclear fuel into the United States and reprocessing it).

To illustrate the level of concern that exists regarding the proposed action, DOE has received letters from the U.S. Department of State, the Nuclear Regulatory Commission, the Arms Control and Disarmament Agency, the International Atomic Energy Agency, and the foreign research reactor operators, all urging DOE to resume acceptance and management of foreign research reactor spent nuclear fuel. Failure to manage this spent nuclear fuel would encourage international commerce in HEU.

HEU fuel in research reactors is more of a proliferation concern than the plutonium in these reactors for three reasons.

First, it is much easier to fashion a simple nuclear weapon out of HEU than plutonium. The chemical separation of HEU metal from HEU spent nuclear fuel would be simpler than the chemical separation of plutonium metal from LEU spent nuclear fuel. HEU is also less radioactive than plutonium, so it presents less of a health hazard to the people working with it. Furthermore, pure plutonium metal is extremely pyrophoric; that is, small chips of pure plutonium metal can ignite spontaneously unless the metal is handled very carefully in special facilities. All these problems with plutonium make HEU fuel a more attractive target for diversion into a nuclear weapons program.

Second, the amount of HEU that would be removed from current civil commerce under the proposed action is much greater than the amount of plutonium that would be produced in replacement LEU fuel elements in the same reactors. The proposed action involves the removal of approximately 4.6 metric tons (5.1 tons) of HEU. For comparison, the plutonium that would be produced in the replacement LEU nuclear fuel is so dilute that even if all the plutonium were somehow extracted, only about 120 kg (265 lb) of plutonium would be produced. This reduction in the amount of weapons grade material in circulation in the future would significantly reduce the threat of nuclear proliferation.

Third regarding fresh (not spent) nuclear fuel containing HEU, this nuclear material would be ideal for diversion into a nuclear weapons program because it would not require chemical separation as spent nuclear fuel would. In the absence of a policy to eliminate HEU from civil commerce, fresh HEU fuel would be shipped to the foreign research reactors from foreign suppliers such as Russia or China. Such shipments would present an additional proliferation risk that would not exist if research reactors worldwide operate on the LEU fuel cycle.

If the United States takes no action to accept foreign research reactor spent nuclear fuel, or otherwise eliminate much of the HEU it contains (e.g., by blending it with natural or depleted uranium to make LEU), one or more of the following events is almost certain to occur. First, some of the foreign research reactors would simply shut down. This does not solve the concerns regarding foreign research reactor spent nuclear fuel, however, since the countries whose reactors were forced to shut down could argue, rightly or wrongly, that the United States was not living up to its obligations under the *Treaty on the Non-Proliferation of Nuclear Weapons* to assist nonnuclear weapons States in the peaceful application of nuclear energy. In addition, the spent nuclear fuel already discharged by the foreign research reactors would still be in storage at the reactor sites. For about 70 percent of the foreign research reactors, this spent nuclear fuel would contain HEU that could be diverted into the production of nuclear weapons if it were removed from the spent nuclear fuel.

Second, some of the foreign research reactors might continue operating and allow their inventory of spent nuclear fuel to build up, much of it containing HEU. The United States could, theoretically, assist such countries (and those with shutdown reactors still holding spent nuclear fuel) in building modern, diversion-resistant spent nuclear fuel storage facilities. This could be extremely expensive, as there are approximately 104 research reactors located in 41 foreign countries. Furthermore, even if perfectly secure storage facilities were built, all that would be required to frustrate their function would be a coup or other change in government leaving a regime in power that is unconcerned about the proliferation of nuclear weapons. Then the spent nuclear fuel could be diverted into a weapons production program, despite the storage assistance that had been provided by the United States. It seems clear that the potential for such an event to occur would increase with the number of spent nuclear fuel stockpiles that are allowed to build up around the world and the length of time they exist.

Finally, some foreign research reactors would be likely to reprocess their spent nuclear fuel. Much of the U.S.-origin enriched uranium was exported under agreements that require prior consent by the United States before the spent nuclear fuel can be shipped to another country, as would be required for almost all reprocessing of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. However, approximately 50 percent of the foreign research reactor spent nuclear fuel is located in countries where such prior consent is not required. The current practice of the most likely reprocessing plant (i.e., the facility in Dounreay, Scotland) is to allow the customer (i.e., the foreign research reactor operator) to specify the form of the separated uranium and its disposition. Thus, the foreign research reactor operator could specify that any separated HEU should be returned as HEU. Furthermore, neither Dounreay nor any other reprocessing facility currently accepts or has the capability to reprocess the high-density LEU fuels that the United States is encouraging foreign research reactors to use to replace the HEU fuels. Thus, in the absence of action to resolve the questions of the disposition of spent nuclear fuel, outlined above, any foreign research reactor operator that reprocesses to control the inventory of spent nuclear fuel must continue to use, or convert back to, fuel containing HEU. Reprocessing under these circumstances leads to perpetuation of the HEU fuel cycle.

While there is some danger of diversion of the HEU in the spent nuclear fuel while it is in storage, the threat is relatively low since the uranium is an integral part of the solid metal spent nuclear fuel elements and is mixed with highly radioactive fission products. A sophisticated chemical processing plant would be required to separate the uranium and convert it into a form suitable for use in a nuclear weapon. However, once the uranium has been separated from the spent nuclear fuel in a reprocessing plant, the situation is fundamentally changed. Any HEU separated in a reprocessing plant would be readily usable for nuclear weapons production, essentially the same as fresh HEU that has never been in a reactor. Therefore, despite U.S. confidence in the capability and determination of the United Kingdom and France to properly safeguard any separated HEU in their reprocessing plants, once the uranium leaves their facilities, the potential for illegal diversion of the material during transit or in the country of destination increases markedly. The rate of increase for potential diversion depends both on the countries and international waterways through which the material must be shipped, the country of final destination, and the number and size of the shipments being made.

Although it is unlikely that all the HEU that has been exported can be recovered or blended down to LEU through the proposed action, it is in the best interest of the United States to make every effort to ensure the proper management of the largest fraction of this material. Arrangements would have to be worked out with foreign reprocessors that would be supportive of U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

By proposing a policy for management of certain foreign research reactor spent nuclear fuel, DOE and the Department of State do not seek to indefinitely accept or otherwise manage spent nuclear fuel from foreign research reactors. Rather, the purpose of the proposed new policy is to remove as much U.S.-origin HEU as possible from international commerce while giving the foreign research reactors and their host countries time to convert to operation with LEU fuel and make their own arrangements for disposition of subsequently generated LEU spent nuclear fuel. The foreign research reactor operators and countries in which the research reactors are operating must be prepared to implement their own arrangements for disposition of their spent nuclear fuel after the policy expires.

1.3 Scope of the Environmental Impact Statement (EIS)

This EIS evaluates the potential environmental impacts that could result from the DOE and Department of State joint proposal to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed policy. The purpose of the proposed policy is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and ultimately eliminating, HEU from civilian commerce. This EIS identifies and evaluates the potential environmental impacts of management alternatives to the proposed action. Implementation of Management Alternative 1 to the proposed action could include the receipt of foreign research reactor spent nuclear fuel at one or more U.S. marine ports of entry, overland transport to one or more DOE sites, and management (interim storage and ultimate disposition) in the United States. DOE will also analyze near-term chemical separation as an alternative to storing the intact spent nuclear fuel pending ultimate disposition.

Under Management Alternative 1 to the proposed action, DOE would accept or otherwise manage spent nuclear fuel containing HEU and LEU from 41 foreign nations, if such spent nuclear fuel is already discharged[4] or would be discharged within the 10-year policy period. Figure 1-1 displays the geographic locations of these nations. The United States would bear the full cost for the management of the foreign research reactor spent nuclear fuel from developing nations.[5] For developed nations, however, the United States would charge a fee for spent nuclear fuel management activities conducted by the United States.[6]

DOE recognizes that Figure 1-1 lists nations that would currently not be considered to be nuclear weapons proliferation risks. History indicates that the United States cannot predict today, with assurance, which countries may develop into proliferation risks in the future. On the other hand, there are good reasons for accepting spent nuclear fuel from nations that are in accord with U.S. nuclear nonproliferation goals, and that have stable governments and excellent nonproliferation credentials. First, several of these countries manufacture research reactors for sale to third world countries. If the United States refuses to help these countries with the management of the spent nuclear fuel from their research reactors, several of their reactors are likely to convert to use of HEU for fuel. If that occurs, their customers in the third world countries would probably also demand to be supplied with reactors fueled with HEU.

Second, both the Soviet Union and China also exported research reactors in the past. The United States is currently engaged in discussions to convince Russia (as the successor supplier to the Soviet Union's allies) and China to work with their nuclear fuel customers to convert their research reactors from HEU to LEU.

---

4 *"Discharged" refers to removal of irradiated fuel from a reactor.*

5 *Developing nations are defined by the World Development Report as having other than high-income economies (World Bank, 1994).*

6 *For purposes of determining which nations are eligible for assistance from the United States in handling their foreign research reactor spent nuclear fuel, Taiwan is considered to be in the high-income economy category.*

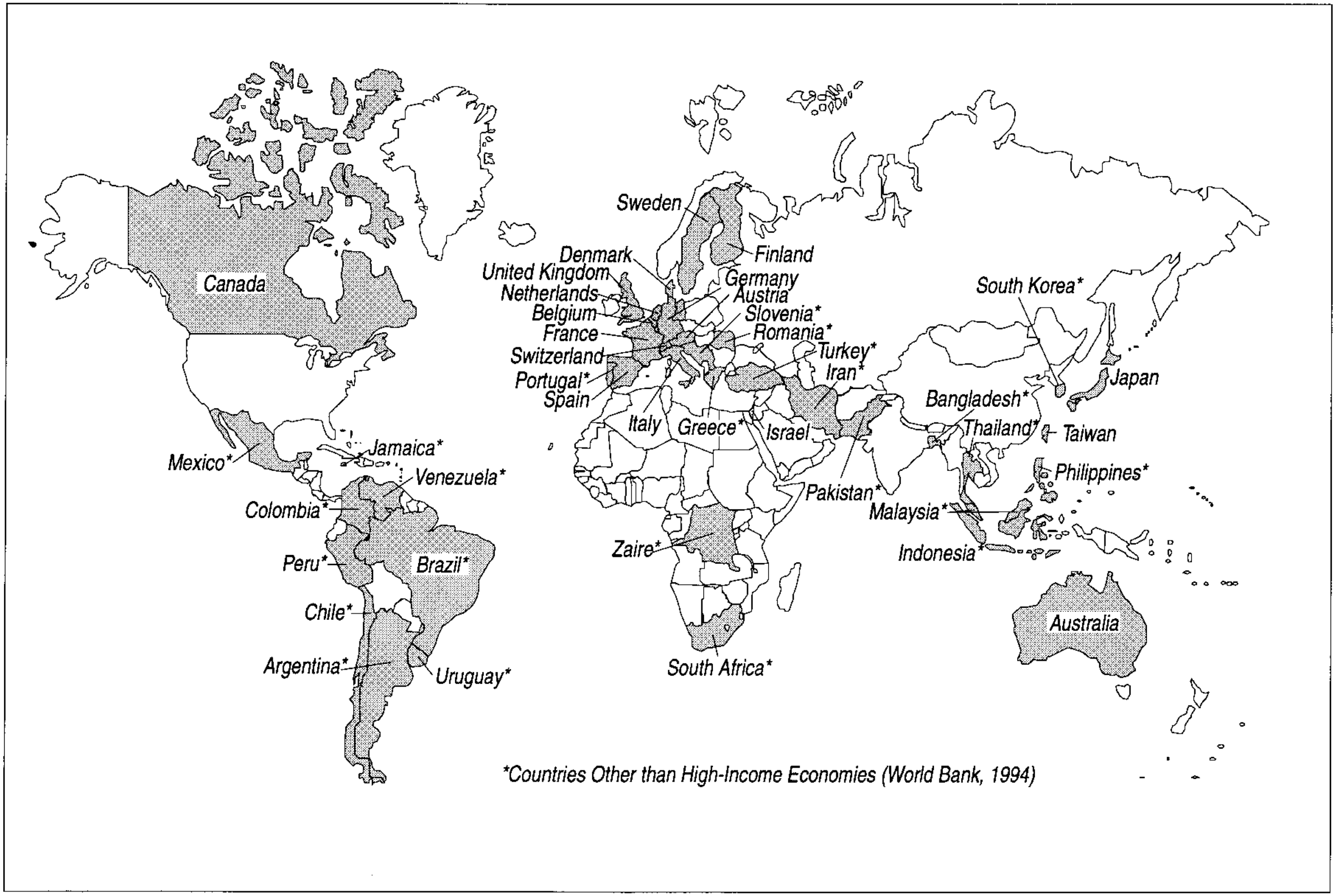


Figure 1-1 Nations with Research Reactors that are Holding or are Expected to Generate Spent Nuclear Fuel Containing Uranium Enriched in the United States

If the United States were to take action that caused its nuclear fuel customers to reconvert their research reactors to HEU, that would likely ensure the failure of U.S. efforts to get Russia and China to encourage their nuclear fuel customers to switch to LEU.

A second alternative (Management Alternative 2) for implementation of the proposed action has been identified and evaluated in this EIS. This alternative includes two subalternatives: (1) to provide assistance to foreign nations with storage of their spent nuclear fuel overseas, and (2) to provide non-technical assistance (financial and/or logistical) in the reprocessing of their spent nuclear fuel. The third alternative (Management Alternative 3) would consist of a combination of various elements of Management Alternatives 1 and 2. For example, a portion of the spent nuclear fuel could be managed overseas, and the remaining portion could be managed in the United States. A No Action Alternative is also evaluated in this EIS.

Under any of these action alternatives, no definitive proposals can be specified at this time for management of the foreign research reactor spent nuclear fuel beyond a 40-year interim management period because insufficient data are available to allow future management proposals to be defined, or for the potential environmental impacts of the final disposition of spent nuclear fuel to be evaluated. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, as well as disposal of vitrified high-level waste resulting from chemical separation of foreign research reactor spent nuclear fuel.

Certain potential actions discussed in this EIS will depend on decisions to be made under other National Environmental Policy Act (NEPA) analyses. Specifically, the site at which the foreign research reactor spent nuclear fuel would be managed, if accepted in the United States, will be selected based on the analyses documented in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (Programmatic SNF&INEL Final EIS) (DOE, 1995c). An exception to this would occur if the "No Action" alternative, or any other alternative that does not include acceptance of foreign research reactor spent nuclear fuel, were selected after completion of the Programmatic SNF&INEL Final EIS. In that case, any site at which foreign research reactor spent nuclear fuel management activities might be conducted in the United States would be selected pursuant to the analyses in this EIS.

The Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995c) was issued on May 30, 1995. In accordance with this Record of Decision, all of the aluminum clad foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory. This is why the Comment Response Document (Volume 3) focuses on the Savannah River Site and Idaho National Engineering Laboratory as sites where any spent fuel accepted in the United States under the proposed policy would be managed, consistent with the Programmatic SNF&INEL Final EIS Record of Decision. Nevertheless, all five of the spent nuclear fuel management sites originally considered in the Draft EIS have been kept in this Final EIS to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL EIS (DOE, 1995c and 1994m).

In this EIS, DOE and the Department of State, in consultation with other government agencies, designate the acceptance and management of foreign research reactor spent nuclear fuel and target material in the United States as the preferred alternative.

1.4 Decisions to be Made Based on this EIS

The principal policy decision for which this EIS will provide a basis is whether DOE and the Department of State should adopt a policy for the United States to manage foreign research reactor spent nuclear fuel containing U.S.-enriched uranium. A necessary part of this decision is the amount and types of foreign research reactor spent nuclear fuel to be managed under such a policy.

If a decision is made to adopt such a policy, then decisions on management implementation must also be made. This EIS is intended to provide the necessary analysis, not only for the decision on adoption of a policy, but also for decisions on management implementation of such a policy, if adopted. Should the decision be to manage the spent nuclear fuel in the United States, decisions to be made on implementation of such a policy include:

- the duration of any foreign research reactor spent nuclear fuel management policy;
- the modes of ocean and overland transport required for shipping any foreign research reactor spent nuclear fuel to be accepted into the United States;
- the ports of entry through which DOE would receive foreign research reactor spent nuclear fuel;
- the need for the construction of new facilities or modification of existing facilities to manage the foreign research reactor spent nuclear fuel to be accepted, and the design, construction, and operation of any such new or modified facilities;
- whether to use near-term chemical separation of the foreign research reactor spent nuclear fuel, as an alternative to storing intact foreign research reactor spent nuclear fuel; and
- whether to accept uranium target material that was enriched in the United States and irradiated in foreign research reactors during the production of molybdenum-99 for medical purposes (in addition to the foreign research reactor spent nuclear fuel discussed above).

1.5 Relationship of this EIS to Other NEPA Documents and Reports Relating to Spent Nuclear Fuel Management

The relationship of this EIS to other DOE NEPA reviews, either completed or currently under preparation, and other DOE analyses is discussed in this section. These reviews and analyses are:

1. *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel.* This Environmental Assessment covers marine transport, receipt, overland transport, and interim wet storage in the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site of up to 409 elements of spent nuclear fuel from foreign research reactors. The Environmental Assessment and associated Finding of No Significant Impact were issued on April 22, 1994.

   The proposed action analyzed in this Environmental Assessment was intended to ensure that the eight research reactors from which urgent-relief spent nuclear fuel shipments were proposed would continue to participate in the RERTR program (a key U.S. nuclear weapons nonproliferation program) until this EIS could be completed and a decision made on whether to adopt and implement the proposed policy to manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

2. *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Environmental Impact Statement (Programmatic SNF&INEL EIS).* Volume 1 analyzes at a programmatic level the potential environmental impacts over the next 40 years of alternatives related to the transportation, receipt, processing, and storage of spent nuclear fuel under the responsibility of DOE. This EIS was prepared in compliance with the order of the U.S. District Court for the District of Idaho [*Public Service Company of Colorado v. Andrus*, Memorandum of Opinion (December 22, 1993)]. The Final EIS and the Record of Decision were published on April 28, 1995 and May 30, 1995, respectively. This EIS formed the basis for deciding, on a programmatic level, which sites will be used for the management of the various types of spent nuclear fuel to which DOE holds title. It included the amount of foreign research reactor spent nuclear fuel that might be accepted in its assessment of potential impacts, and addressed the sites at which the foreign research reactor spent nuclear fuel could be stored if a decision is made to accept foreign research reactor spent nuclear fuel. The Record of Decision indicated that aluminum clad spent fuel will be consolidated at the Savannah River Site and non-aluminum clad fuel will be managed at the Idaho National Engineering Laboratory. On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a Consent Order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent fuel to Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such foreign research reactor spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to Idaho National Engineering Laboratory under the stipulated settlement and Consent Order. Notwithstanding the Record of Decision, a full analysis of all five management sites considered in the Programmatic SNF&INEL Final EIS has also been included in this EIS to maintain maximum consistency with the analysis provided in the Programmatic SNF&INEL EIS.

3. *Waste Management Programmatic EIS.* The Waste Management Programmatic EIS has evolved from the formerly named Environmental Management Programmatic EIS, as was most recently described in a *Federal Register* Notice (55 FR 42633), on August 22, 1990. The Draft Waste Management Programmatic EIS was issued in August 1995 and analyzes programmatic alternatives for DOE-wide management of five waste types: high-level radioactive waste, low-level radioactive waste, low-level mixed waste, transuranic waste, and hazardous waste. DOE expects to issue the Final Waste Management Programmatic EIS in late 1996.

4. *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and their Environmental Safety and Health Vulnerabilities.* The Spent Fuel Working Group Report, dated November 1993, presents a comprehensive assessment of the conditions of spent nuclear fuel and other irradiated materials stored at DOE facilities. Eight DOE sites were identified as containing storage facilities with major vulnerabilities that need to be resolved. The vulnerabilities identified in this report have been considered in the analysis of the actions that would be involved in management of foreign research reactor spent nuclear fuel in the United States.

5. *Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities.* The Plan of Action to Resolve Spent Nuclear Fuel Vulnerabilities is a three-phased approach to remedy vulnerabilities identified in the Spent Fuel Working Group Report. The Phase-I Plan of Action, dated

February 1994, addressed 31 of 33 high-priority vulnerabilities and 48 lower-priority issues. The Phase-II Plan of Action, dated April 1994, was the product of follow-on work to the Phase-I report, and resolved a majority of the funding issues associated with spent fuel vulnerabilities. The Phase-III Plan of Action, issued in October 1994, focused on the resolution of critical policy issues and provided individual action plans that addressed all the identified vulnerabilities. In the preparation of this EIS on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel, the actions being taken to resolve these issues have been evaluated to ascertain that any existing facilities considered for use in receipt, handling, or storage of foreign research reactor spent nuclear fuel are capable of performing satisfactorily. The EIS evaluates storage capabilities and alternatives at the interim storage sites and whether potential storage sites would be capable of immediately implementing the proposed action.

6. *Interim Management of Nuclear Materials Final Environmental Impact Statement.* This EIS considers the impacts of managing nuclear materials stored at the Savannah River Site, including stabilization by separating the nuclear materials from fission and decay products in the chemical separation facilities and conversion of the resulting liquids to solids in waste and material conversion facilities, including the Defense Waste Processing Facility, the FB-, HB, and FA-Lines, and the saltstone facility. The Final EIS was issued in October 1995. A Record of Decision and Notice of Preferred Alternative was published in the *Federal Register* (60 FR 65300) on December 19, 1995. Decisions were made for the majority of materials covered by the EIS in the Record of Decision and processing Mark-16 and Mark-22 fuels and blending down the resulting HEU to LEU was identified as the preferred alternative for those materials. These fuels are similar to the aluminum-based foreign research reactor spent nuclear fuel although significant corrosion has been identified. An amended Record of Decision is expected soon regarding the Mark-16 and Mark-22 fuels. DOE has taken and will take into consideration all Records of Decision on the Interim Management of Nuclear Materials Final EIS in the preparation of this EIS and in reaching a decision on how to implement the proposed policy, if adopted.

7. *Disposition of Surplus Highly Enriched Uranium Environmental Impact Statement.* The Draft EIS, issued in October 1995, assesses the environmental impacts of alternatives for the disposition of U.S.-origin HEU that has or may be declared surplus to national defense and defense-related program needs, in order to eliminate the nuclear proliferation risk and, where practical, recover economic value and peaceful, beneficial reuse of the material. Under the preferred alternative, the HEU would be blended to LEU at four sites, including the Savannah River Site; under this alternative, most of the HEU would be blended to LEU as fuel feed for commercial nuclear power plants to generate electricity, while that which cannot meet commercial fuel specifications would be blended to low level waste. The Final EIS and Record of Decision are scheduled for April and May 1996, respectively.

8. *Storage and Disposition of Weapons-Usable Fissile Materials Programmatic Environmental Impact Statement.* This EIS will assess the environmental impacts of reasonable alternatives for safe, secure and internationally-accountable long-term storage of non-surplus, U.S.-origin plutonium and HEU, long-term storage of plutonium and HEU that are not part of the strategic reserves need for weapons research and development, and post-interim storage of surplus fissile material prior to disposition. This programmatic EIS will also analyze reasonable alternative strategies and technologies for disposition of U.S.-origin plutonium that is or may be declared surplus to defense and defense-related program needs,

in order to eliminate the proliferation risk by making the material as proliferation-resistant as spent fuel. The Savannah River Site will be analyzed in conjunction with the various alternatives for both storage and disposition. The Notice of Intent to prepare a programmatic EIS was issued in June 1994 (59 FR 31985), and the scope of the EIS was revised in April 1995 (60 FR 17344). The Draft EIS is expected in early 1996, and the Final EIS and Record of Decision are expected in late 1996.

9. *Environmental Impact Statement Evaluating Container Systems for the Management of Spent Nuclear Fuel.* This EIS was originally titled *Environmental Impact Statement for a Multi-Purpose Canister System for the Management of Civilian and Naval Spent Nuclear Fuel.* This EIS, as described in 59 FR 53442 (1994), was intended to address the potential environmental impacts associated with alternative systems for storage and transport of spent nuclear fuel assemblies for civilian spent nuclear fuel. DOE decided for programmatic reasons in November 1995 to withdraw its proposal to prepare this EIS. The Department of the Navy has announced in a *Federal Register* Notice (60 FR 62829) that it will take the lead in preparing this EIS for evaluating container systems for the management of Navy spent nuclear fuel. DOE is a cooperative agency on this EIS.

10. *F-Canyon Plutonium Solutions Environmental Impact Statement.* This EIS evaluated the potential environmental impacts over the next 10 years of alternatives for stabilization of plutonium solutions currently stored in the F-Canyon at the Savannah River Site. The plutonium solutions remain from reprocessing operations that DOE suspended in 1992 at the Savannah River Site. The Record of Decision for this EIS announced that DOE would implement the preferred alternative analyzed in the EIS. This alternative is to process the plutonium solutions to plutonium metal.

11. *Defense Waste Processing Facility Supplemental Environmental Impact Statement.* This Supplemental EIS examines the cumulative environmental impacts of modifications made to the Defense Waste Processing Facility and associated high-level waste facilities at the Savannah River Site since the issuance of the 1982 EIS. The preferred alternative for the proposed action under this Supplemental EIS is to continue construction and begin operation of the Defense Waste Processing Facility as designed. The Final Supplemental EIS was completed in November 1994 and the Record of Decision was issued on April 12, 1995 (60 FR 18589). The DOE decision was to complete construction and startup testing and begin operation of the facility as currently designed. One of the Implementation Alternatives considered in the Foreign Research Reactor Spent Nuclear Fuel EIS is to chemically separate a portion of the foreign research reactor spent nuclear fuel at the Savannah River Site and vitrify the resulting high-level radioactive waste in the Defense Waste Processing Facility.

12. *Tritium Supply and Recycling Programmatic Environmental Impact Statement.* This Programmatic EIS evaluated the siting, construction and operation of tritium supply technology alternatives and the recycling facilities at five candidate DOE sites. The EIS also evaluated the use of a commercial reactor for producing tritium. Currently, DOE does not have the capability to produce tritium in the required amounts. The Savannah River Site in South Carolina, which will receive the aluminum-based foreign research reactor spent nuclear fuel, has been identified by DOE as the preferred site for an accelerator, should one be constructed, and the site for the upgrade and consolidation of existing recycling facilities.

The Final Programmatic EIS was completed and issued to the public in October 1995. The DOE Record of Decision was issued on December 12, 1995 (60 FR 63891) with a decision to implement the preferred alternatives.

13. *Stockpile Stewardship and Management Programmatic Environmental Impact Statement.* This EIS was originally a part of the Nuclear Weapons Complex Reconfiguration Programmatic Environmental Impact Statement. The Notice of Intent of this EIS was published on June 14, 1995 (60 FR 31291) after a prescoping workshop on May 19, 1995. This Programmatic EIS will examine activities required to maintain a high level of confidence in the safety and reliability of a reduced stockpile of nuclear weapons in the absence of underground nuclear testing. The Savannah River Site at Aiken, South Carolina, which houses the tritium loading/unloading and surveillance of tritium reservoirs, will receive the aluminum-based foreign research reactor spent nuclear fuel should the proposed action be implemented in the United States. This draft EIS is expected to be issued for public comment in February 1996.

1.6 Structure of this EIS

The remainder of this EIS is structured as follows:

- Chapter 2 presents the proposed action, describes management alternatives for implementation of the proposed action, alternative means of implementing each management alternative, and a No Action Alternative. Chapter 2 specifies the Preferred Alternative that has been developed by DOE and the Department of State.
- Chapter 3 characterizes the affected environments at potential ports of entry and at potential foreign research reactor spent nuclear fuel management locations.
- Chapter 4 addresses the policy considerations and the potential environmental impacts of each management alternative for implementation of the proposed action, alternative means of implementing each management alternative, and a No Action Alternative.
- Chapter 5 describes the international and domestic regulations governing radioactive materials that apply to DOE actions that might be taken under this EIS.
- Chapters 6, 7, and 8 contain primarily reference information, such as the List of Preparers, Agencies Consulted, and References, respectively.

The appendices to this document present details of the evaluations and analyses performed for this EIS.

2. Proposed Action and Alternatives

This chapter states the proposed action and describes the management alternatives analyzed in this Environmental Impact Statement (EIS) for implementation of the proposed action. Environmental and policy impacts from the management alternatives are presented in Chapter 4.

2.1 Overview of the Proposed Action and Alternatives

The U.S. Department of Energy (DOE) and Department of State are jointly proposing to adopt a policy to manage spent nuclear fuel from foreign research reactors. Only spent nuclear fuel containing uranium enriched in the United States would be covered by the proposed action. The purpose of the proposed action is to promote U.S. nuclear weapons nonproliferation policy objectives, specifically by seeking to reduce, and eventually eliminate, highly-enriched uranium (HEU) from civilian commerce.

To implement the proposed action, DOE and the Department of State have considered three foreign research reactor spent nuclear fuel management alternatives. They are:

1. To accept and manage foreign research reactor spent nuclear fuel in the United States (Management Alternative 1);

2. To facilitate the management of foreign research reactor spent nuclear fuel at one or more foreign locations (Management Alternative 2); and

3. A combination of elements from Management Alternatives 1 and 2 (Management Alternative 3, Hybrid Alternative).

The management alternatives of the proposed action are portrayed in Figure 2-1 and are discussed in more detail in Sections 2.2, 2.3, and 2.4.

A No Action Alternative on the part of DOE and the Department of State to address the status of the foreign research reactor spent nuclear fuel has also been considered in this EIS. The No Action Alternative is discussed in Section 2.5.

DOE and the Department of State have identified a preferred alternative for the proposed action. The preferred alternative is described in Section 2.9.

The foundation for the analysis presented in Chapter 4 of this EIS is the evaluation of the components that comprise the basic implementation of Management Alternative 1. The basic implementation concept is an attempt by DOE and the Department of State to avoid unnecessary repetition by selecting a reasonable option for each component and examining them in detail under Management Alternative 1. Since the No Action Alternative would not have any direct environmental impacts in the United States, it requires only policy analysis in this EIS. Management Alternatives 1, 2, and 3, however, would all have environmental impacts in the United States, and the components of the basic implementation provide the parameters with which to analyze their potential environmental impacts in this EIS.

The detail of analysis provided for the basic implementation components is based on the fact that some variation of these components is utilized in each implementation alternative under Management Alternative 1, as well as in Management Alternative 3. In this way, analysis of the implementation

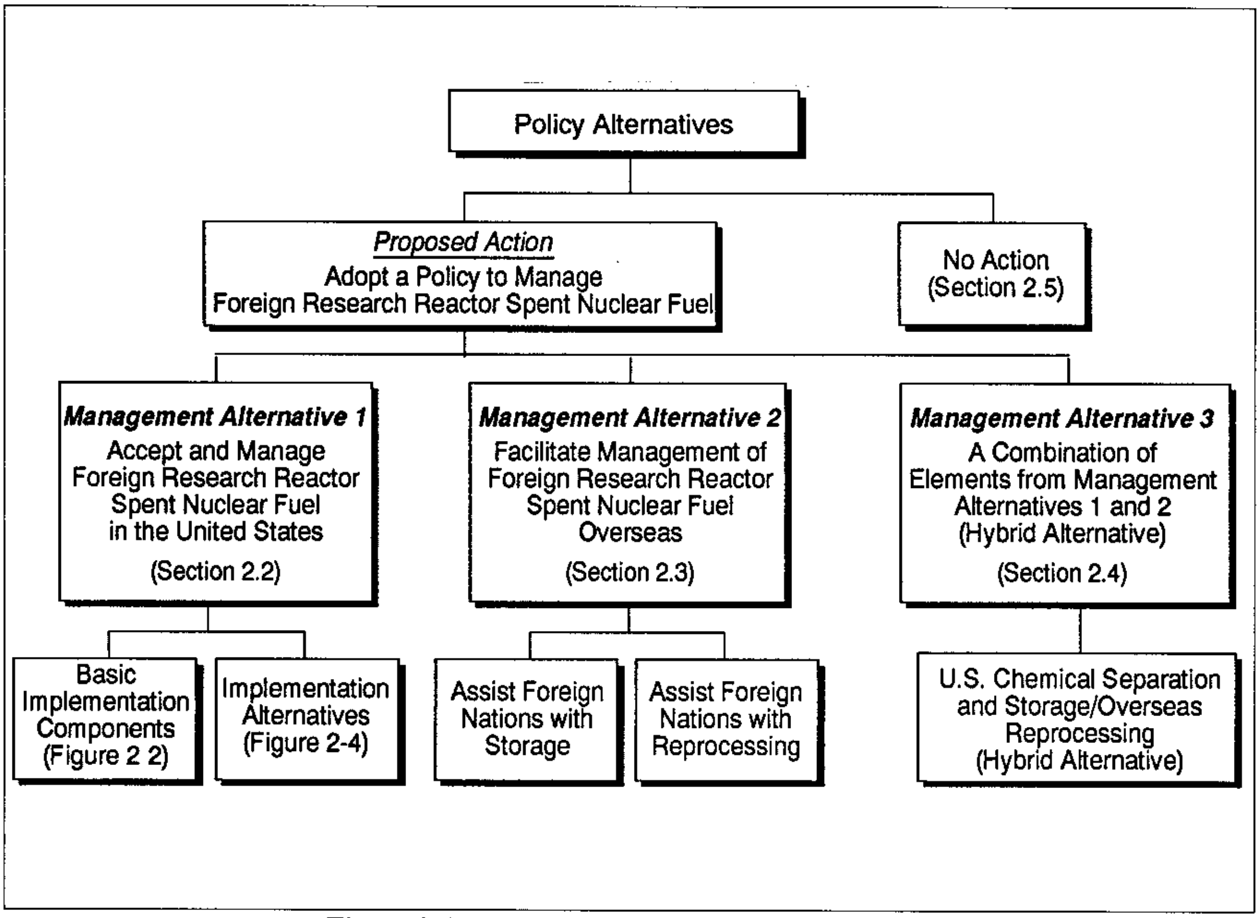


Figure 2-1 Policy and Management Alternatives

alternatives, as well as Management Alternative 3, can be tiered from the analysis of the basic implementation. In and of itself, the basic implementation of Management Alternative 1 is a viable implementation alternative for consideration under Management Alternative 1, along with the other implementation alternatives discussed below. However, the level of detail contained in the analysis of the basic implementation does not indicate any preference for this alternative. Rather, it merely eliminates the need to duplicate information later in the analysis.

The components of the basic implementation of Management Alternative 1 would consist of the following:

1. A policy duration of 10 years.

2. A financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries a competitive fee.

3. The receipt of a fixed amount of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. This fixed amount is up to approximately 22,700 foreign research reactor spent nuclear fuel elements and is based on estimated inventories of foreign research reactor spent nuclear fuel currently stored or to be generated in the 10-year policy period.

4. Taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit (19 km or 12 mi), or continental U.S. borders for shipments from Canada.

5. Marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships.

6. Ports of entry that qualify on the basis of criteria discussed in this EIS.

7. Ground transport from ports of entry to storage sites, and between sites (by truck, rail, or barge, or a combination of these modes.)

8. Potential storage sites identified in the Programmatic SNF&INEL Final EIS (DOE, 1995c) for foreign research reactor spent nuclear fuel, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

9. Use of dry storage technology for construction of new storage facilities.

The basic implementation components are depicted in Figure 2-2 and described in Section 2.2.1. Environmental impacts and policy considerations of the basic implementation components of Management Alternative 1 are presented in Section 4.2.

Utilizing the components provided above, DOE has evaluated seven implementation alternatives for Management Alternative 1 in addition to the basic implementation. Each implementation alternative is comprised of the same components as the basic implementation; however, for the purpose of analysis, one of the components has been varied. The seven implementation alternatives are given below.

1. Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation.

2. Acceptance of foreign research reactor spent nuclear fuel for a period of time different from the time period identified in the basic implementation.

3. Financial arrangements different from those identified in the basic implementation.

4. Taking title to foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation.

5. Use of wet storage technology for construction of new storage facilities instead of dry storage technology as identified in the basic implementation.

6. Use of near term conventional chemical separation in the United States to reduce the duration of, and amount of, spent nuclear fuel storage required.

7. Use of new developmental treatment and/or packaging technologies in addition to storage as identified in the basic implementation.

Implementation alternatives for Management Alternative 1 are discussed in Section 2.2.2. The environmental impacts and policy considerations of the implementation alternatives are discussed in Section 4.3.

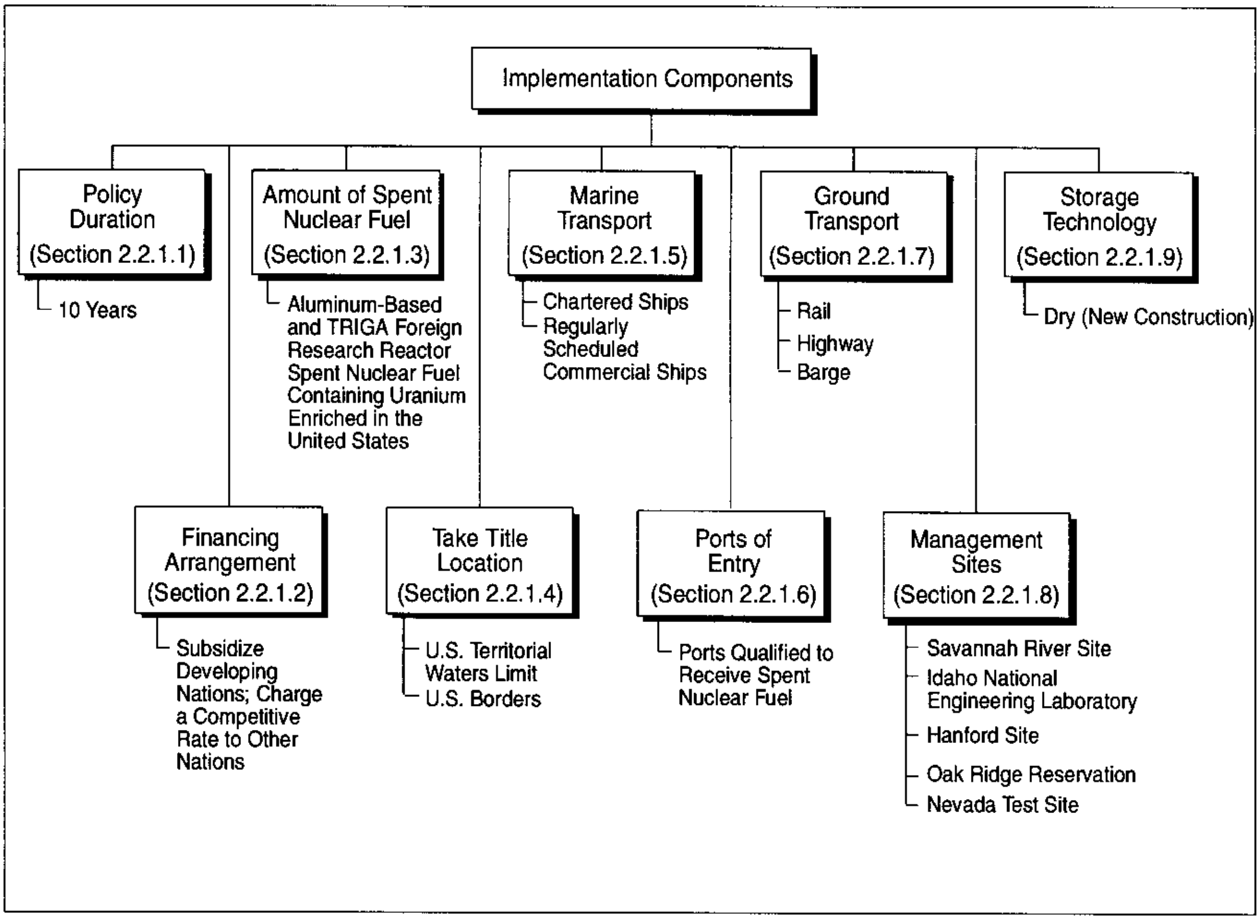


Figure 2-2 Basic Implementation Components

Qualifying Fuel Types and Policy Stipulations

This policy applies solely to aluminum-based and TRIGA[1] research reactor fuels and target materials containing HEU and low enriched uranium (LEU) of U.S. origin. Aluminum-based fuel is clad in aluminum and has an active fuel region that consists of an alloy of uranium and aluminum or a dispersion of uranium-aluminide, uranium-oxide[2] or uranium-silicide in aluminum. TRIGA fuel consists of an alloy of uranium and zirconium and is clad in either aluminum or stainless steel. Fuels containing significant quantities of Uranium-233 (^{233}U) are excluded. Target materials are the residual materials from isotope production targets in research reactors.

The policy would include the following stipulations:

- Spent nuclear fuel (either and/or both HEU and LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.

1 *TRIGA stands for Training, Research, Isotope, General Atomic reactors.*

2 *This uranium-oxide composition refers to aluminum-clad fuel plates or tubes containing dispersions of U_3O_8 in aluminum. It does not include fuels containing UO_2 pellets clad in aluminum, zircaloy, stainless steel, or other materials, or uranium-silicide in aluminum.*

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.

- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.

- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.

- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).

- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such spent nuclear fuel). However, since the repository characterization program is in its early stages, the waste acceptance criteria for deposit of DOE's spent nuclear fuel in a repository have not been developed. Thus, a determination cannot be made at this time as to the requirements that must be met to allow placement of the foreign research reactor spent nuclear fuel in the repository. As a result, the EIS analysis for the time period beyond 40 years is qualitative rather than quantitative. The qualitative assessment includes consideration of disposal of intact foreign research reactor spent nuclear fuel, disposal of vitrified high-level waste resulting from chemical separation, as well as utilization of various potential new technologies to process the spent nuclear fuel into a more stable form prior to its ultimate disposition. In the event that the geologic repository schedule is delayed beyond the 40-year program period, DOE would continue to manage the foreign research reactor spent nuclear fuel or any resultant stable waste forms in existing facilities at the DOE management site(s) until a geologic repository becomes available. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

2.2 Management Alternative 1 - Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

2.2.1 Basic Implementation Components

2.2.1.1 Policy Duration

The policy duration of the basic implementation of Management Alternative 1 would be 10 years, beginning on the date when the management policy becomes effective. Spent nuclear fuel containing HEU and LEU of U.S. origin that is currently being stored or is to be generated during the 10-year period of the policy would be accepted.

Actual shipments of spent nuclear fuel to the United States could be made for a period of 13 years starting from the effective date of the policy implementation, as long as the spent nuclear fuel was generated within the 10-year policy period. The 3 additional years would allow for a cooling time for fuel discharged from a reactor late in the policy period, logistics in arranging for shipment of this fuel, as well as other unplanned for potential delays.

2.2.1.2 Financing Arrangements

The United States would bear the full cost of transporting and managing the foreign research reactor spent nuclear fuel received from developing countries. Developing countries are defined by the World Bank as those countries having other than high-income economies (World Bank, 1994). For developed countries, however, the United States would charge a competitive fee for the handling, storage, conditioning (as needed), and any disposal activities conducted by the United States. Tables 2-1 and 2-2, which provide estimates of the number of elements that may be accepted, identify those countries defined as developing countries.

2.2.1.3 Amount of Foreign Research Reactor Spent Nuclear Fuel

The analysis in this EIS is based primarily on the number of individual elements of foreign research reactor spent nuclear fuel that could be accepted. When appropriate, the analysis also uses two other measures to express the amounts in understandable terms:

- Mass of Heavy Metal. This is the mass of all the heavy metal atoms in the spent nuclear fuel (mostly uranium), excluding the mass of other materials such as alloys, cladding, and structural materials. The international standard unit of measure for this quantity is metric tons of heavy metal (MTHM).
- Volume. The volume of the spent nuclear fuel is important because it determines the number of shipments and the storage space required. The volume is expressed in cubic meters.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation is up to approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The number of elements cited for acceptance under the policy includes those elements at issue in the Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994m).

Table 2-1 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (MTHM)*[a]	*Estimated Number of Shipments*
Argentina[b]	283	0.071	9
Australia	975	0.427	9
Austria	157	0.191	5
Belgium	1,766	0.730	59
Brazil[b]	155	0.099	5
Canada	2,831	4.478	116
Chile[b]	58	0.012	2
Colombia[b]	16	0.002	1
Denmark	660	0.529	22
France	1,962	3.442	149
Germany	1,504	0.909	49
Greece[b]	239	0.113	8
Indonesia[b]	198	0.236	6
Iran[b]	29	0.006	1
Israel	192	0.111	6
Italy	150	0.043	5
Jamaica[b]	2	0.001	1
Japan	2,981	3.134	99
Korea (South)[b]	168	0.321	7
Netherlands	1,488	1.404	49
Pakistan[b]	82	0.016	3
Peru[b]	29	0.039	1
Philippines[b]	50	0.024	2
Portugal[b]	88	0.054	3
South Africa[b]	50	0.010	2
Spain (from Scotland)[c]	40	0.016	1
Sweden	1,113	1.374	37
Switzerland	159	0.128	5
Taiwan	127	0.066	4
Thailand[b]	31	0.005	1
Turkey[b]	69	0.089	2
United Kingdom	12	0.004	1
Uruguay[b]	19	0.018	1
Venezuela[b]	120	0.082	4
Total	17,803	**18.184**	**675**

[a] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[b] *Countries with other than high-income economies (World Bank, 1994).*

[c] *40 Spent nuclear fuel elements of Spain's JEN-1 Reactor core are stored in Dounreay, Scotland.*

Implementation of Management Alternative 1 would involve less than 1 percent of the total mass of heavy metal that DOE currently manages as spent nuclear fuel (DOE, 1994c), and approximately 10 percent of the volume.

Table 2-2 Estimated Number of TRIGA[a] Reactor Spent Nuclear Fuel Elements Generated by Foreign Research Reactor Operators by January 2006

Country	Estimated Number of Spent Nuclear Fuel Elements	Initial Mass of Uranium (MTHM)[b]	Estimated Number of Shipments
Austria	106	0.020	3
Bangladesh[c]	100	0.049	3
Brazil[c]	75	0.014	3
Finland	171	0.033	6
Germany	358	0.068	12
Indonesia[c]	245	0.047	8
Italy	386	0.072	13
Japan	326	0.062	11
Korea (South)[c]	336	0.064	11
Malaysia[c]	94	0.047	3
Mexico[c]	186	0.035	6
Philippines[c]	128	0.079	4
Romania[c]	1,451	0.189	48
Slovenia[c]	393	0.075	13
Taiwan	144	0.086	5
Thailand[c]	136	0.035	4
Turkey[c]	79	0.015	2
United Kingdom	90	0.017	3
Zaire[c]	136	0.026	4
Total	**4,940**	**1.033**	**162**

[a] *TRIGA is an acronym for Training, Research, Isotope, General Atomic reactors.*

[b] *To derive uranium mass in kilograms, multiply the amounts by 1,000.*

[c] *Countries with other than high-income economies, developing countries (World Bank, 1994)*

The Notice of Intent [59 Fed. Reg. 54336 (1993)] for this EIS estimated that 15,000 spent nuclear fuel elements would be accepted under the proposed action. This estimate [representing 12 MTHM, with a volume of approximately 89 m^3 (3,200 ft^3)] was prepared in early 1993, based on a projected 10-year period of generation of spent nuclear fuel at foreign research reactors in 28 foreign countries, plus the spent nuclear fuel available at these foreign research reactors as of 1993.

Since preparation of the 1993 spent nuclear fuel projection, however, cooperative understandings have been reached with several other foreign research reactor operators concerning their participation in the proposed spent nuclear fuel management program. In addition, the period of time over which the management policy would be in effect has been delayed by 3 years (to mid-1996) and thus at least 3 more years' worth of spent nuclear fuel has accumulated. Thus, the amount of spent nuclear fuel from foreign research reactors that would be accepted under the basic implementation is increased to a new total of up to 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 spent nuclear fuel elements of the type considered in the 1993 projection. Of this amount, approximately 4.6 MTHM is HEU, and 14.6 MTHM is LEU foreign research reactor spent nuclear fuel.

Tables 2-1 and 2-2 provide an estimate of the amount of spent nuclear fuel that has been or would be generated in each country by late 2005 (10 years from the effective date of the policy implementation), as estimated by Argonne National Laboratory based on information provided by the foreign research reactor operators (Matos, 1994). A list of the foreign research reactors included in the proposed policy is provided in Appendix B. Table 2-1 shows the inventory of aluminum-based fuel clad in aluminum, while Table 2-2 shows zirconium-based TRIGA fuel clad in either aluminum or stainless steel. These two tables are combined to yield the approximately 22,700 elements (or about 19.2 MTHM) that are estimated to be currently stored or generated by the year 2005. The tables also provide the estimated number of shipments expected from each country. The number of shipments is a key parameter in evaluating the risks associated with the handling and transportation of the foreign research reactor spent nuclear fuel.

It should be noted that the number of elements and number of shipments presented for each country in Tables 2-1 and 2-2 are estimates based on projections of the numbers of elements to be generated over a ten-year period into the future. These estimates are intended to conservatively bound the total number of foreign research reactor spent nuclear fuel elements and shipments associated with the proposed policy. However, the actual distribution of elements and shipments among the listed countries might change, within the limits of the total number of elements and shipments listed, based on actual experience gained during the lifetime of any policy that may be established.

For the purpose of analysis, the foreign research reactor spent nuclear fuel has been categorized by fuel type (aluminum-based or TRIGA) and geography (Eastern or Western) depending on the location of the likely port(s) of entry to the United States. As noted in Section 2.6.4.1, foreign research reactor spent nuclear fuel from Europe, Africa, and the Middle East and parts of Central and South America is likely to enter the United States from the east coast (Eastern) and the rest from the west coast (Western).

The distribution of the foreign research reactor spent nuclear fuel under the basic implementation is as follows:

- By Fuel Type: Aluminum-based — approximately 17,800 elements, 18.2 MTHM, 105 m^3 (3,900 ft^3)

  TRIGA — approximately 4,900 elements, 1.0 MTHM, 5 m^3 (200 ft^3)

- By Geography: Eastern — approximately 16,400 elements, 14.4 MTHM, 80 m^3 (3,000 ft^3)

  Western — approximately 6,300 elements, 4.8 MTHM, 30 m^3 (1,100 ft^3)

The assumptions used in estimating the number of shipments are included in Appendix B (Section B.1.6). Characteristics of the foreign research reactor spent nuclear fuel that would be accepted are provided in Section 2.6.1.

2.2.1.4 Location for Taking Title to Foreign Research Reactor Spent Nuclear Fuel

Under the basic implementation of Management Alternative 1, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters, or continental U.S. borders for shipments from Canada. Where DOE takes title would not have an effect on the environment. Title location of the spent nuclear fuel is relevant to questions that include the source and extent of liability for damage in the event of an accident outside the scope of Price-Anderson Act coverage. The Price-Anderson Act [42 U.S.C. §2210 (1988)] provides a mechanism by which DOE could pay for damages arising out of a nuclear incident that occurs within the United States.

2.2.1.5 Marine Transport

Under the basic implementation of Management Alternative 1, the foreign research reactor spent nuclear fuel would be shipped by chartered and/or regularly scheduled commercial ships from foreign ports to the United States. Chartered shipments would be on purpose-built ships or general purpose commercial cargo ships meeting appropriate International Marine Organization regulations. Regularly scheduled commercial shipments would be on general purpose commercial ships carrying other cargo at the same time.

Marine transport of the foreign research reactor spent nuclear fuel, as well as ground transport between ports and management sites in the United States, would be carried out in approved and certified spent nuclear fuel casks. These casks would be certified for use in the United States by the U.S. Department of Transportation if the cask was designed and fabricated in a foreign country, or by the U.S. Nuclear Regulatory Commission (NRC) if the cask was designed and fabricated in the United States. The design and fabrication of casks in a foreign country is based on the International Atomic Energy Agency standards which are the bases for those promulgated by the NRC. Marine transport activities are discussed in more detail in Section 2.6.3.

All ships entering U.S. territorial waters are required to comply with U.S. Coast Guard safety regulations and are subject to U.S. Coast Guard inspection. In addition, international transportation of hazardous material is governed by the International Movement of Dangerous Goods Code, which is one of a series of safety codes associated with the International Maritime Organization. This code establishes the international rules for shipping hazardous cargos, which includes foreign research reactor spent nuclear fuel. While most nations have agreed to follow the International Maritime Organization codes, including the International Movement of Dangerous Goods Code, compliance by individual shippers would be voluntary.

Unless DOE were to take title to the foreign research reactor spent nuclear fuel overseas (Section 2.2.2.4), the responsibility for shipping the spent nuclear fuel to the United States (if the fuel is to be accepted into the United States) belongs to the foreign research reactor operators. Under these conditions, DOE would ensure that the shipment of the spent nuclear fuel was accomplished on well-equipped, -maintained, and -operated ships through the contract that would be signed between DOE and every participating foreign research reactor operator. DOE would require the use of carriers that commit to following the International Movement of Dangerous Goods Code and all other safety requirements, such as the Safety of Life at Sea, through these contracts. If DOE were to be responsible for shipping, only shipping firms that guaranteed to follow U.S. Coast Guard regulations and international safety codes would be used to ship foreign research reactor spent nuclear fuel.

2.2.1.6 Port(s) of Entry

The basic implementation of Management Alternative 1 would involve receipt of foreign research reactor spent nuclear fuel at any of the 10 ports of entry chosen on the basis of criteria discussed in Section 2.6.3.1. All 10 candidate ports offer standard cargo container unloading services. These potential ports of entry have been identified subsequent to the application of criteria, (including appropriate experience, safe transit, adequate facilities, and population) to the universe of potential U.S. marine ports of entry. These ports are: Charleston, SC (includes Naval Weapons Station [NWS] Charleston and Wando Terminal); Galveston, TX; Hampton Roads, VA (includes Newport News, Norfolk, and Portsmouth terminals); Jacksonville, FL; Military Ocean Terminal Sunny Point (MOTSU), NC; NWS Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA; and Wilmington, NC. The geographic location of each of these ports is displayed in Figure 2-3. This EIS will also assess the potential impacts of foreign

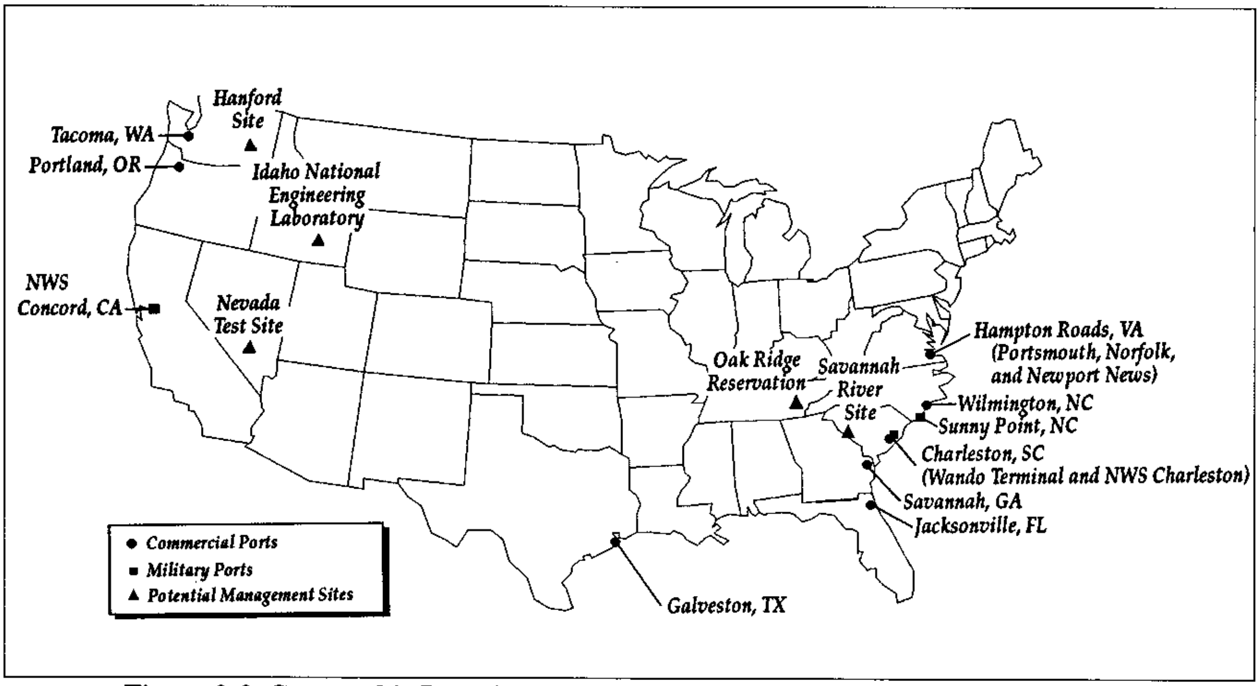


Figure 2-3 Geographic Locations of the Ports of Entry Considered for Receipt of Foreign Research Reactor Spent Nuclear Fuel

research reactor spent nuclear fuel at three high-population-density ports to bound the results of the impact analysis. These high-population-density ports are: Elizabeth, NJ; Long Beach, CA; and Philadelphia, PA. The port identification and evaluation process is discussed in Section 2.6.3 and in Appendix D.

2.2.1.7 Ground Transport

The basic implementation of Management Alternative 1 would involve shipment of foreign research reactor spent nuclear fuel from the ports of entry (both seaports and Canadian border crossings) to potential management sites. It could also involve shipment of foreign research reactor fuel between management sites. As explained in Section 2.6.4.1, the unavailability of certain sites to accept foreign research reactor spent nuclear fuel at the beginning of the management policy period would necessitate temporary receipt and management of foreign research reactor spent nuclear fuel at an available site and subsequent transportation to another site. The ground transport options and route identification process are discussed in Section 2.6.4.

Both rail and highway shipping capabilities are available at all ports of entry and each management site under consideration, with the exception of the Nevada Test Site, which has no rail capability. The shipment of foreign research reactor spent nuclear fuel was analyzed along representative highway and railway routes between all ports and the potential management sites as applicable. Barge transportation is also considered where applicable. The only management sites reasonably accessible by barge are the Savannah River Site and the Hanford Site from the ports of Savannah, GA and Portland, OR, respectively.

2.2.1.8 Foreign Research Reactor Spent Nuclear Fuel Management Sites

Potential sites considered by DOE for the receipt and management of foreign research reactor spent nuclear fuel under this EIS are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). They are the Savannah River Site, the Idaho National Engineering Laboratory, the

Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. Site-specific activities associated with the basic implementation of Management Alternative 1 and the implementation alternatives are discussed in Section 2.6.5. There are only two sites in the United States that could start receiving these shipments quickly: the Savannah River Site and the Idaho National Engineering Laboratory.

2.2.1.9 Storage Technologies

Under the basic implementation of Management Alternative 1, DOE would receive and manage foreign research reactor spent nuclear fuel for a period starting in approximately mid 1996, and continuing for 40 years until ultimate disposition. During the first few years, storage would take place in existing storage facilities that use both wet and dry storage technologies. For the period beyond those first few years, when construction of new facilities may become necessary, the storage technology identified for the basic implementation of Management Alternative 1 is dry storage. However, construction of new wet storage facilities is considered as an implementation alternative. Storage technologies and storage facilities considered under the basic implementation of Management Alternative 1 and implementation alternatives are discussed in Section 2.6.5.

2.2.2 Implementation Alternatives for Management Alternative 1

Environmental effects of each of the implementation alternatives are evaluated in this EIS. The range of these alternatives is presented in Figure 2-4. Chapter 4, Section 4.3, provides results of the analysis.

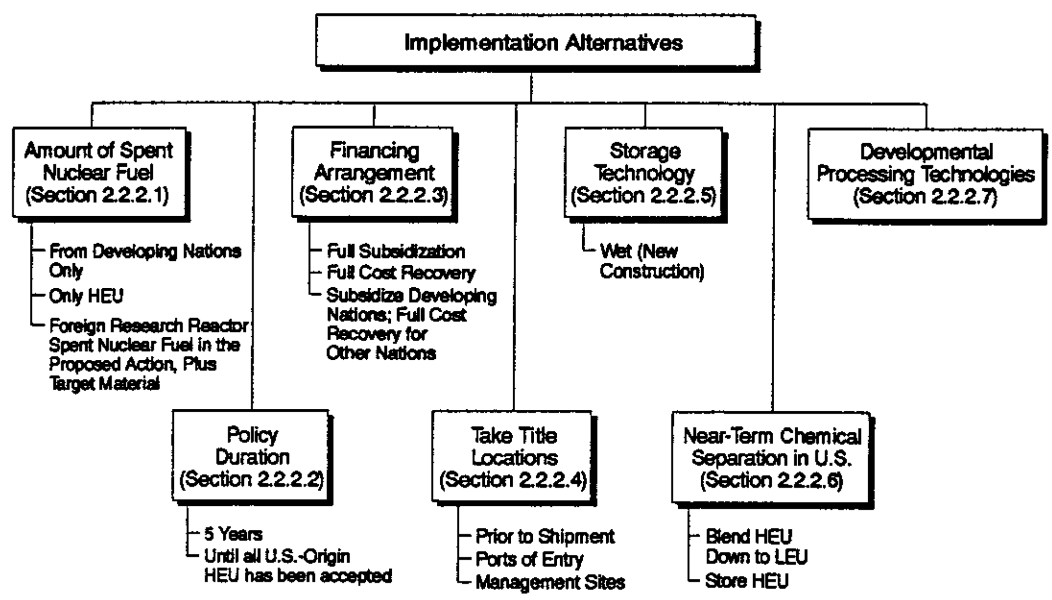


Figure 2-4 Implementation Alternatives

2.2.2.1 Implementation Alternative 1 - Alternative Amounts of Spent Nuclear Fuel to be Accepted

This implementation alternative involves choosing to accept and manage one of three subalternative amounts of foreign research reactor spent nuclear fuel:

1a. Accept spent nuclear fuel (HEU and/or LEU) only from developing countries. The foreign research reactor spent nuclear fuel from these countries contains approximately 1.9 MTHM, representing 5,000 individual elements, with a volume of 13 m^3 (500 ft^3).

1b. Accept only HEU from the research reactors eligible under the proposed action. The amount of this HEU would be approximately 4.6 MTHM, representing 11,200 elements, with a volume of 61 m^3 (2,250 ft^3).

1c. In addition to foreign research reactor spent nuclear fuel, accept HEU and LEU target materials that were used in Canada, Belgium, Argentina, and Indonesia for the production of medical isotopes. Isotope production targets[3] are irradiated in research reactors and dissolved in acid or base to extract radioisotopes that are used in medical imaging applications. The residual materials after dissolution and extraction of the radioisotopes are referred to here as target material. It is expected that this target material would contain about 0.6 MTHM, representing the uranium content of approximately 620 typical foreign research reactor spent nuclear fuel elements.

Under the last subalternative, HEU target material would be accepted until a suitable LEU target is available. After such a time, target material would be accepted from a foreign research reactor only if that foreign research reactor agrees to convert to use of LEU target.

2.2.2.2 Implementation Alternative 2 - Alternative Policy Durations

The basic implementation of Management Alternative 1 has a duration of 10 years. Two policy duration subalternatives were assessed. These are:

2a. *Five-Year Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, spent nuclear fuel (HEU and LEU) currently stored or generated during the 5-year policy period would be accepted.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or having life time cores, or planning to shut down by a specific date while the policy is in effect, or for which a suitable LEU fuel is not available, HEU fuel currently stored or generated during the 5-year policy period would be accepted.

- For foreign research reactors that are already shut down, spent nuclear fuel (HEU and LEU) currently stored would be accepted.

3 *Canada, Argentina, and Belgium currently use aluminum-based targets containing HEU, and Indonesia currently uses a target that consists of a layer of HEU oxide material plated on the interior surface of a stainless steel tube.*

The amount of spent nuclear fuel estimated to be accepted for a 5-year policy period under this subalternative is up to approximately 18,800 individual elements containing approximately 13 MTHM, with a volume of 87 m^3 (3,300 ft^3). The distribution by fuel type and geography is as follows:

- By Fuel Type: Aluminum-based — approximately 14,100 elements; 12 MTHM, 83 m^3 (3,100 ft^3).

  TRIGA — approximately 4,700 elements, 1.0 MTHM, 4 m^3 (200 ft^3).

- By Geography: Eastern — approximately 13,400 elements, 9.5 MTHM, 65 m^3 (2,400 ft^3)

  Western — approximately 5,400 elements, 3.4 MTHM, 22 m^3 (900 ft^3)

This subalternative would allow shipments and receipt of foreign research reactor spent nuclear fuel to be made for 8 years starting from the effective date of the policy implementation, as long as the fuel had been generated within the 5-year policy period. The additional 3 years would allow for a cooldown time of fuel discharged late in the 5-year period, the logistics in arranging shipment of this fuel, as well as other possible delays from strikes, court actions, and mechanical problems.

2b. *Indefinite HEU/10-Year LEU Policy:*

- For foreign research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective, LEU spent nuclear fuel currently stored or generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel that had been or would be discharged from the reactor would continue indefinitely, until all such HEU spent nuclear fuel had been accepted.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and agreeing to convert to LEU fuel, or planning to shut down by a specific date within 10 years of the effective date of the policy, LEU spent nuclear fuel generated in the 10-year policy period would be accepted within the time period allowed in the basic implementation (13 years). Acceptance of HEU spent nuclear fuel would continue indefinitely, until all such HEU spent nuclear fuel had been received.

- For foreign research reactors operating on HEU fuel when the policy becomes effective and for which a suitable LEU fuel is not available, or having life time cores, HEU spent nuclear fuel would be accepted:

  - from foreign research reactors with lifetime cores, until all such HEU spent nuclear fuel had been accepted, and

  - from other foreign research reactors until all HEU spent nuclear fuel at the reactor on the date the policy becomes effective, or generated within 5 years of that date, had been accepted.

- For foreign research reactors that are already shut down, the LEU spent nuclear fuel would be accepted within the period allowed in the basic implementation. The HEU spent nuclear fuel would be accepted indefinitely, until all such HEU spent nuclear fuel had been accepted.

Under this implementation subalternative, the total amount of foreign research reactor spent nuclear fuel that would be accepted is the same as in the basic implementation of Management Alternative 1.

2.2.2.3 Implementation Alternative 3 - Alternative Financing Arrangements

Under the basic implementation, the costs of participation would be fully subsidized by the United States for developing countries, however, developed countries would be charged a competitive fee.

For this implementation alternative, the cost impacts of the following subalternatives arrangements were evaluated:

- Subsidize all countries;
- Charge all countries the full cost of accepting and managing the foreign research reactor spent nuclear fuel (a full-cost recovery fee); and
- Subsidize developing countries as in the basic implementation, and charge developed countries a full-cost recovery fee.

A full-cost recovery fee would be based on the estimated cost to the United States for the safe, final disposition of the spent nuclear fuel within the United States. This fee could be based on: (1) the cost of chemically separating spent nuclear fuel and disposal of vitrified high-level waste, or (2) interim storage of the spent nuclear fuel followed by direct ultimate disposition.

In theory, this arrangement would cost the United States nothing. All costs would be borne by the foreign nations. However, many developing, and some developed nations probably would decline to pay these high costs, which could lead to HEU spent nuclear fuel stockpiled around the world, much of it remaining in the countries least able to protect it. For many countries, this arrangement would have the same impact as the No Action Alternative.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation, or under the implementation subalternatives discussed in Section 2.2.2.1. The actual amount of spent nuclear fuel received could be less than that identified under the basic implementation because, as stated above, some countries may consider the higher fee to be a disincentive. The analysis of environmental impacts for this EIS, however, considered the amount of spent nuclear fuel to be received for this implementation alternative to be unchanged for use as an upper bounding case.

2.2.2.4 Implementation Alternative 4 - Alternative Locations for Taking Title

In the basic implementation, DOE would take title to the foreign research reactor spent nuclear fuel at the limit of U.S. territorial waters (19 km or 12 mi), or the continental U.S. border for shipments from Canada. The location for taking title is relevant to questions of liability and regulatory authority. For example, if DOE were to take title at the foreign research reactor site, there could be additional regulatory burdens on DOE, due to the laws of a particular country being imposed upon the owner of the spent nuclear fuel. The taking of title prior to shipment might impose upon DOE additional legal liability for damages not associated with a nuclear incident covered by the Price-Anderson Act. DOE and the Department of State considered the following three subalternative approaches regarding the locations for taking title to the foreign research reactor spent nuclear fuel:

- Taking title to the foreign research reactor spent nuclear fuel before shipment,
- Taking title at the port(s) of entry, and
- Taking title at the management site(s).

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.5 Implementation Alternative 5 - Wet Storage Technology for New Construction

Under the basic implementation, storage requiring new construction would employ dry storage technology. As an implementation alternative, DOE has assessed the use of wet storage technologies for new construction, which use water-filled pools to store spent nuclear fuel. Wet storage methods have been used historically at DOE sites and by the nuclear industry.

The amount of spent nuclear fuel that would be accepted under this implementation alternative would be the same as that identified under the basic implementation or under the implementation subalternatives discussed in Section 2.2.2.1.

2.2.2.6 Implementation Alternative 6 - Near Term Conventional Chemical Separation in the United States

Under this implementation alternative, near term conventional chemical separation would be conducted at either the Savannah River Site or the Idaho National Engineering Laboratory. There are both advantages and disadvantages to chemical separation of foreign research reactor spent nuclear fuel. The advantages include the following:

- The high-level radioactive waste from the foreign research reactor spent nuclear fuel would be transformed into forms that are more suitable (i.e., more compact and stable) for storage than intact aluminum-based spent nuclear fuel.
- The high-level waste would be converted to a form that is expected to be acceptable for disposal in a geologic repository.
- Construction of some or all of the new spent nuclear fuel storage space would be avoided.
- The conventional chemical separation facilities already exist, as well as the waste treatment facilities required to put the high-level radioactive and other wastestreams in forms suitable for disposal. In contrast, there are the large technical, cost, and regulatory uncertainties associated with direct disposal of intact foreign research reactor spent nuclear fuel (much of it containing HEU).
- If disposal of intact spent nuclear fuel is shown to be technically infeasible, or if the waste acceptance criteria for a geologic repository require significant dilution of the HEU due to criticality concerns, DOE estimates that the life-cycle costs of chemical separation may be substantially lower than the cost of storage and geologic disposal of intact spent nuclear fuel. (Alternatively, if direct disposal of intact foreign research reactor spent nuclear fuel,

including that containing HEU, is shown to be technically feasible, DOE estimates that the costs of chemical separation and the storage/direct disposal option would be nearly the same.)

The disadvantages include the following:

- Chemical separation would increase the total volume of the waste (including liquid high-level waste raffinates, transuranic wastes, various solid and liquid low-level wastestreams, acidic wastes, chelating and complexing agents, and solvents). Volume reduction and other treatments would be used to prepare these wastes for disposal. (Because the requirements for direct disposal of aluminum-based spent nuclear fuel have not been established, the character and volumes of waste associated with direct disposal are uncertain.)

- The separated uranium, which DOE would prefer to blend down to LEU, would have to be stored until it could be sold or otherwise disposed of.

- The forms of the wastes generated by chemical separation are complex, involving corrosive, flammable and toxic liquids.

- The use of chemical separation by the United States as a spent nuclear fuel management technology could increase the accumulation of stockpiles of HEU unless the HEU is blended down. The United States does not engage in chemical separation for nuclear explosive purposes, and seeks to eliminate, where possible, the accumulation of stockpiles of HEU or plutonium. The United States nuclear weapons nonproliferation policy on reprocessing is summarized in the White House Fact Sheet on Nonproliferation and Export Control Policy dated September 27, 1993. A copy of this policy is included in Appendix G of this EIS.

Taking these advantages and disadvantages into account, chemical separation of foreign research reactor spent nuclear fuel in existing facilities is not preferred by DOE as a technology for routine management of spent nuclear fuel in the United States. Nonetheless, chemical separation remains a reasonable alternative in light of DOE's substantial technical experience in these operations and the availability of existing facilities.

DOE is considering the development and use of various alternatives to chemical separation for foreign research reactor spent nuclear fuel stabilization, interim storage and conditioning for disposal under Management Alternative 1, Implementation Alternative 7 in this EIS. This initiative is discussed in more detail in a DOE memorandum of December 28, 1994 from Thomas P. Grumbly to Jill E. Lytle (see Appendix G). In fact, development, demonstration and implementation of a new treatment and/or packaging technology is a key element of the preferred alternative as described in Section 2.9.

The Nuclear Waste Policy Act of 1982 (as amended) authorizes disposal of the foreign research reactor spent nuclear fuel in a geologic repository (if DOE takes title to such fuel). The foreign research reactor spent nuclear fuel, and/or the high-level radioactive waste that would result from chemical separation of the foreign research reactor spent nuclear fuel, would require storage until its disposal in such a geologic repository. A determination of whether the foreign research reactor spent nuclear fuel can be safely disposed of in a geologic repository will depend on the outcome of scientific analyses, including a repository performance analysis considering the final form in which the foreign research reactor spent nuclear fuel would be emplaced in the repository.

Near term conventional chemical separation of foreign research reactor spent nuclear fuel at the other three potential foreign research reactor spent nuclear fuel management sites was not analyzed because the Oak Ridge Reservation and the Nevada Test Site do not have facilities in which such chemical separation could be conducted, and the facilities at the Hanford Site are no longer operable. To consider chemical separation at any of these three sites, the foreign research reactor spent nuclear fuel would need to be stored in the United States during the period of time that a new chemical separation facility at one of these sites was designed, a project-specific NEPA review was conducted, and the facility constructed and put into operation. Such activities could not be completed in the near term and, accordingly, these sites were not considered reasonable alternatives for near term chemical separation.

Solid low-level radioactive waste and wastewater generated by chemical separation would be managed in the same manner as the similar wastes generated by the storage of the intact foreign research reactor spent nuclear fuel. Discussion of waste generation from storage is included in Appendix F, Section F.4. Chemical separation would also generate five types of waste that would not result from storage of intact foreign research reactor spent nuclear fuel: high-level radioactive waste, hazardous waste, mixed hazardous and radioactive waste, and low-level "saltstone" waste.

Following chemical separation of the foreign research reactor spent nuclear fuel, the resulting high-level radioactive wastes would be managed along with substantial existing inventories of identical waste. Management of the high-level radioactive wastes would include the following:

1. The high-level wastes would be transferred to storage tanks and kept there pending processing;
2. The wastes would be pretreated in preparation for further processing;
3. To preclude the necessity for transporting liquid high-level wastes, these wastes would be processed on the sites where they were generated:
    a. At the Savannah River Site, the wastes would be:
        1) Vitrified in the Defense Waste Processing Facility;
        2) The borosilicate glass resulting from vitrification would be stored pending disposal;
    b. At the Idaho National Engineering Laboratory, the wastes would be:
        1) Calcined to produce a more easily stored waste form;
        2) Stored in the calcine form pending development of a process and facility for final processing;
        3) After further research and development regarding conversion techniques and waste forms, the calcine would be converted to a form suitable for geologic disposal and stored pending disposal;
4. The final waste form would be transported to and disposed of in a geologic repository.

Transuranic wastes[4] would be stored on the site where the chemical separation would be accomplished until a permanent disposal facility, such as the Waste Isolation Pilot Plant, becomes available. Site treatment plans for low-level and transuranic mixed wastes are now being developed. Hazardous wastes would be sent to a licensed commercial treatment, storage and disposal facility. Saltstone, a mixture of low-level waste and concrete that is a by-product of high-level radioactive waste vitrification at the Savannah River Site's Defense Waste Processing Facility, would be pumped into above-ground concrete vaults onsite, where it would harden into a concrete monolith.

The Savannah River Site currently has chemical separation facilities. This capability, however, is limited to aluminum-based spent nuclear fuel. In contrast, the Idaho National Engineering Laboratory has facilities that can chemically separate both aluminum-based and TRIGA foreign research reactor spent nuclear fuel. However, these facilities would require some upgrades in order to accomplish this chemical separation. The existing dissolvers and calcination vessel could be used at the start of chemical separation activities, but would have to be replaced within a few years. A new tank farm and set of calcine bins would have to be built. Furthermore, this site does not have an existing vitrification facility or a glass waste storage building, as the Savannah River Site does. Upgrading the facilities at the Idaho National Engineering Laboratory would require additional time and funding. This EIS analyzes the impacts of chemically separating aluminum-based foreign research reactor spent nuclear fuel at both the Savannah River Site and the Idaho National Engineering Laboratory, but considers chemical separation of TRIGA foreign research reactor spent nuclear fuel only at the Idaho National Engineering Laboratory.

Under the near term chemical separation alternative, there are two components: Extent of the Chemical Separation and Uranium Disposition. Each of these components is discussed below.

Extent of the Chemical Separation

At each of the two potential sites, the foreign research reactor spent nuclear fuel could be chemically separated in a dedicated mode or as part of larger scale chemical separation activities.

Chemical Separation at the Savannah River Site Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would chemically separate all 18.2 MTHM of the aluminum-based foreign research reactor spent nuclear fuel, shown previously in Table 2-1. The Savannah River Site has facilities that could perform the chemical separation, so no new chemical separation facilities would need to be constructed.

DOE and the Department of State have included in this EIS analysis all 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel in Table 2-1 because this is the maximum that could be considered. It is not possible to specify how much of the aluminum-based foreign research reactor spent nuclear fuel might be chemically separated before the chemical separation facilities would have been shut down, so the analysis is based on the entire amount. If chemical separation of all the foreign research reactor spent nuclear fuel were selected, existing chemical separation facilities would be required to remain operational for the 13-year duration of receipts. Maintenance and operation of a facility dedicated solely to chemical separation of foreign research reactor spent nuclear fuel is considered to be an inefficient use of such facilities. Thus, this is a nonpreferred subalternative.

---

4 No transuranic waste would be generated if the transuranic elements (mostly plutonium) were not extracted during chemical separation. The trace amounts of these elements that exist in the foreign research reactor spent nuclear fuel could remain in the high-level wastestream.

Chemical Separation at the Idaho National Engineering Laboratory Dedicated to Foreign Research Reactor Spent Nuclear Fuel: DOE would restart the facilities and chemically separate all the aluminum-based and TRIGA foreign research reactor spent nuclear fuel, shown previously in Tables 2-1 and 2-2. The Idaho National Engineering Laboratory does not have all the facilities required to perform the chemical separation, so some new facilities would need to be constructed. Furthermore, DOE announced a Record of Decision on May 30, 1995 for the Programmatic SNF&INEL Final EIS. Chemical separation at Idaho National Engineering Laboratory was not included in this Record of Decision, so additional site-specific NEPA documentation would be required to restart these chemical separation facilities.

DOE and the Department of State have included in this analysis all 19.2 MTHM of foreign research reactor spent nuclear fuel in Tables 2-1 and 2-2, because this is the maximum that could be considered. It is not possible to specify how much of the foreign research reactor spent nuclear fuel might be chemically separated in this case, so the analysis is based on the entire amount. Up to approximately 12 years of operation would be required to chemically separate this amount. The construction and operation of new facilities for chemical separation dedicated to foreign research reactor spent nuclear fuel is considered inefficient, and therefore, this subalternative is not preferred.

Chemical Separation at the Savannah River Site as Part of Larger Scale Activities: DOE is in the process of preparing other NEPA reviews and making decisions that could affect the decisions to be made in this EIS. The Interim Management of Nuclear Materials Final EIS (DOE, 1995a) analyzed alternatives for stabilization of nuclear materials currently stored at the Savannah River Site that represent health and safety risks, as stored in their current forms and locations. The nuclear materials in the Interim Management of Nuclear Materials Final EIS that most closely resemble the aluminum-based foreign research reactor spent nuclear fuel are the Mark-16 and Mark-22 fuels. The preferred alternative for these fuels, as announced in the *Federal Register* (60 FR 65300), is chemical separation. Therefore, the near term chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site as part of larger scale activities is predicated upon a decision by DOE to use the chemical separation facilities at the Savannah River Site to chemically separate the Mark-16 and Mark-22 fuels under the Interim Management of Nuclear Materials Final EIS.

The Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE announced in its Record of Decision on May 30, 1995 that it intends to consolidate all its aluminum-based spent nuclear fuel at the Savannah River Site. DOE could also chemically separate other aluminum-based spent nuclear fuel that is transported to the Savannah River Site under this EIS.

The aluminum-based foreign research reactor spent nuclear fuel shown in Table 2-1 would be chemically separated along with other DOE aluminum-based spent nuclear fuel selected for chemical separation. This could amount to a maximum of 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 0.56 MTHM of target material under Implementation Subalternative 1c (Section 2.2.2.1), approximately 28.8 MTHM of other aluminum-based spent nuclear fuel currently stored at the Savannah River Site (Wichmann, 1995), and approximately 3.4 MTHM of aluminum-based spent nuclear fuel that could be transported to the Savannah River Site (Wichmann, 1995). In all, about 51 MTHM could be chemically separated at the Savannah River Site, requiring up to approximately 13 years of operation. The environmental impacts analysis is presented in Section 4.3.6.

Chemical Separation at the Idaho National Engineering Laboratory as Part of Larger Scale Activities: Volume 2 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) includes a brief analysis of the alternative of restarting the chemical separation facilities at the Idaho National Engineering Laboratory for stabilization of nuclear materials. These facilities are currently being cleaned up in preparation for decommissioning, so restarting them would require additional site-specific NEPA documentation, but DOE is not currently performing NEPA analysis on restarting these facilities.

Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c) considered the alternative site(s) where DOE's spent nuclear fuel (including foreign research reactor spent nuclear fuel) would be managed. DOE could chemically separate other spent nuclear fuel that is transported to the Idaho National Engineering Laboratory under this EIS.

All the aluminum-based spent nuclear fuel would be chemically separated along with all the TRIGA foreign research reactor spent nuclear fuel and certain other spent nuclear fuel from the Idaho National Engineering Laboratory. The amount of aluminum-based spent nuclear fuel would be approximately 51 MTHM, as described above. The additional spent nuclear fuel would include 1.0 MTHM of TRIGA foreign research reactor spent nuclear fuel and approximately 13 MTHM of TRIGA and zirconium-based spent nuclear fuel from onsite facilities (Cottam, 1995). In all, about 65 MTHM could be chemically separated under this program at Idaho National Engineering Laboratory, requiring up to approximately 12 years of operation. The environmental impact analysis is presented in Section 4.3.6.

Uranium Disposition

Chemical separation, as the name implies, would separate the uranium from the waste products. The separated LEU could be sold to the commercial sector for reuse as reactor fuel. The HEU disposition issue is being considered in a separate DOE NEPA document, the Disposition of Surplus Highly Enriched Uranium EIS. If DOE decides to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also be blended down. Conversely, if DOE decides not to blend down the HEU that is within the scope of the Disposition of Surplus Highly Enriched Uranium EIS, then the HEU that would be recovered in this implementation alternative would also not be blended down. Until this decision is made, however, DOE may decide to blend down HEU in specific instances. For example, DOE recently announced its decision (60 FR 65300), to blend down the HEU solutions at the Savannah River Site under the Interim Management of Nuclear Materials Final EIS.

The options for HEU disposition at the Savannah River Site are:

1. Blending it down to less than 20 percent enrichment inside the chemical separation facilities;
2. Blending it down to less than two percent enrichment inside the chemical separation facilities and then processing it to an oxide in the existing FA-Line at the Savannah River Site; and
3. Completing construction of the Uranium Solidification Facility at the Savannah River Site, then processing the HEU directly to an oxide, followed by storage in a safe, secure facility.

The options for HEU disposition at Idaho National Engineering Laboratory are:

1. Blending it down to less than 20 percent inside the chemical separation facilities, and
2. Processing the HEU directly to an oxide.

Some minor modifications to the facility would be necessary to blend down the HEU. No modifications would be necessary to process it directly to an oxide. For either option, additional NEPA documentation would be required.

Under any of the blending down options, the nuclear weapons nonproliferation goal would be satisfied because the material would not be usable in weapons after it was blended down. This material could be returned to the commercial sector and reused in nuclear reactors.

If the HEU were not blended down, DOE and the Department of State might be accused of accepting the foreign research reactor spent nuclear fuel in order to stockpile HEU for future weapons use. To address this concern, DOE and the Department of State would, over time, place the separated HEU under International Atomic Energy Agency safeguards. DOE and the Department of State have identified three possible means of implementing this International Atomic Energy Agency safeguards initiative which would require the use of a finite storage area (Material Balance Area) subject to inspections by the International Atomic Energy Agency. These are:

1. DOE could use the only available Material Balance Area: Vault 16 at the Y-12 Plant on the Oak Ridge Reservation. This vault's capacity is about 40 MTHM of HEU and it presently contains only about 10 MTHM, so the available capacity is about 30 MTHM. Chemical separation of all the spent nuclear fuel in this implementation alternative would recover less than 25 MTHM of HEU, so there is sufficient capacity in Vault 16 to store all this material.

2. A new Material Balance Area could be set up at the Savannah River Site or the Idaho National Engineering Laboratory.

3. The HEU could be stored in existing vaults, and brought out to a temporary Material Balance Area for each International Atomic Energy Agency inspection.

If a decision is made to chemically separate the foreign research reactor spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

2.2.2.7 Implementation Alternative 7 - Developmental Treatment and/or Packaging Technologies

Under this implementation alternative, DOE and the Department of State would initiate a development program that could lead to a decision to construct and operate a new facility for treatment and/or packaging of foreign research reactor spent nuclear fuel. The purpose of this potential new facility would be to treat, package and store the foreign research reactor spent nuclear fuel in a manner suitable for geologic disposal, without necessarily separating the fissile materials. Other goals of the development process would be to define a technology and facility that would operate safely, meet or exceed all applicable environmental requirements (including minimization of waste volumes, toxicity, and mobility), and be consistent with U.S. nuclear weapons nonproliferation policies.

There are numerous technologies that DOE and the Department of State could consider under such a development program. These technologies could be applied at any one of the five potential foreign research reactor spent nuclear fuel management sites, and most would require the construction of totally new facilities (although some could be implemented through modifications to existing facilities). The potential environmental impacts of the construction and operation of such a new facility cannot be estimated at this time because the technologies are still developmental and the hypothetical new facility has not been designed. Implementation of any of these technologies would require additional NEPA analysis and documentation, including additional opportunities for public review and comment.

Many of the potentially applicable technologies are already being considered under the DOE Office of Spent Fuel Management Technology Integration Working Group. A development program, such as the one that would be implemented under this alternative, is outlined in the DOE Spent Nuclear Fuel

Technology Integration Plan (DOE, 1994c). A number of these developmental technologies have progressed beyond initial technical feasibility studies and have reasonably defined cost and schedule estimates for further development. Technologies, such as the Plasma Arc Treatment Process and the Electrometallurgical Treatment Process, are being developed by the Pacific Northwest Laboratory and the Argonne National Laboratory, respectively. Other developmental technologies, however, require additional evaluation prior to undergoing detailed development efforts. Some of the development technologies and criticality prevention techniques are briefly described below. Criticality prevention is discussed in Section 2.6.1.

Developmental Treatment Technologies

Chop and Dilute: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and added to shards of depleted uranium-aluminum alloy to prevent a criticality in the repository. The mixture would have an enrichment of no more than one percent (WSRC, 1994a).

Chop and Poison: The foreign research reactor spent nuclear fuel could be rendered into shards in a mechanical chopper and a neutron poison could be added to prevent a criticality in the repository (WSRC, 1994a). A neutron poison is an element that absorbs neutrons without fissioning, thus preventing a fission chain reaction.

Melt and Dilute: The foreign research reactor spent nuclear fuel could be melted with depleted uranium metal added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Melt and Poison: The foreign research reactor spent nuclear fuel could be melted and a neutron poison could be added to the molten mixture to prevent a criticality in the repository (WSRC, 1994a).

Electrometallurgical Treatment: The foreign research reactor spent nuclear fuel could be dissolved, then the aluminum could be separated from the uranium and fission products in an electrorefiner. DOE has proposed to demonstrate this process as a management option for a variety of DOE-owned spent nuclear fuel. This process would produce a mineral waste form containing most of the fission products and a metal alloy containing the rest of the fission products (DOE, 1994c).

Plasma Arc Treatment: The foreign research reactor spent nuclear fuel would be placed directly into a plasma centrifugal furnace with other material (low-enriched uranium, depleted uranium, and neutron absorbers) where it would be melted and converted into a ceramic material.

Chloride Volatility Treatment: The foreign research reactor spent nuclear fuel could be completely volatized to chlorides. This process is being investigated at the Idaho National Engineering Laboratory and would require about 15 years to develop. Then the uranium could be separated and the fission products could be converted to oxides or fluorides for vitrification (DOE, 1994c).

Glass Material Oxidation and Dissolution System: The foreign research reactor spent nuclear fuel could be converted to a glass form in this single-step process. This process was recently invented at Oak Ridge National Laboratory, and has been demonstrated at the laboratory scale. The foreign research reactor spent nuclear fuel would be melted together with glass frit and all process chemistry would occur in this molten mixture. Then, depleted uranium would be added to resolve criticality concerns before the mixture is poured into canisters (DOE, 1994c).

Dissolve and Dilute: The foreign research reactor spent nuclear fuel could be dissolved in acid and depleted uranium could be added to the solution to reduce the enrichment to no more than one percent to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Dissolve and Poison: The foreign research reactor spent nuclear fuel could be dissolved in acid and a neutron poison could be added to prevent a criticality in the repository. Then the solution would be vitrified (WSRC, 1994a).

Developmental Packaging Technologies

Direct Disposal in Small Packages: This is a variation of the "direct disposal"concept. The foreign research reactor spent nuclear fuel could be packed intact into small waste packages to limit the amount of fissile material in any single package. Neutron poisons would also be packed in the packages in the spaces between the fuel element plates or rods to prevent a criticality in the repository.

Can-in-Canister: The foreign research reactor spent nuclear fuel could be canned, then the cans could be encapsulated in glass inside of canisters. The encapsulation process could be performed in the Defense Waste Processing Facility at the Savannah River Site, using a high-level waste glass. In effect, the cans containing foreign research reactor spent nuclear fuel would displace an equal volume of high-level waste glass inside standard Defense Waste Processing Facility canisters.

2.3 Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Under this management alternative, DOE and the Department of State would seek to encourage and facilitate the management of foreign research reactor spent nuclear fuel overseas in a manner that would be consistent with U.S. nuclear weapons nonproliferation policy. DOE and the Department of State have evaluated the following two subalternatives:

1a. Overseas Storage - Encourage and assist foreign research reactors that are able to store their spent nuclear fuel in facilities in their own countries, or in developing countries, as a step toward the final disposition of the spent nuclear fuel. U.S. assistance would be provided to ensure that appropriate storage technologies, regulations, and safeguards were applied.

In some cases, this subalternative might be implemented by expansion of the storage facilities located at the foreign research reactor sites. However, many foreign research reactor operators are associated with academic institutions with limited budgets, or have building restrictions for site-specific reasons (e.g., no physical space for expansion). Thus, the opportunities for expanded spent nuclear fuel storage at foreign research reactor sites may be limited or even nonexistent. In countries with established nuclear power programs, management might also be provided at the sites where such countries will store their power reactor spent nuclear fuel. Some foreign research reactor operators may also be able to make arrangements for indefinite storage at sites in developing countries. Ultimate disposition of the foreign research reactor spent nuclear fuel would still have to be arranged at the conclusion of the management period. In the meantime, foreign research reactor spent nuclear fuel containing HEU would be stored in up to 41 countries around the world.

1b. Overseas Reprocessing - Provide nontechnical (financial and/or logistical) assistance to foreign research reactors and reprocessors to facilitate reprocessing spent nuclear fuel overseas in facilities operated under international safeguards sufficient to satisfy U.S. nuclear weapons nonproliferation concerns. Wherever possible, the wastes resulting from this reprocessing would be returned to the country in which the spent nuclear fuel was irradiated. If the reprocessing wastes cannot be returned to the country in which the spent nuclear fuel was irradiated, such wastes might be accepted by the United States for storage and ultimate geologic disposal.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

The overseas reprocessing option will be evaluated in terms of whatever is supportive of the U.S. nuclear weapons nonproliferation policy on HEU minimization. For example, factors such as the following would have to be considered:

- An expectation that HEU separated during reprocessing would be blended down to LEU for research reactors which are converting to LEU.
- The foreign reprocessors would provide the capability to reprocess LEU as well as HEU.
- Research reactors would be encouraged to convert to LEU if an LEU fuel exists or is developed that will allow such operation.

Arrangements would have to be worked out with foreign reprocessors that would be supportive of U.S. nuclear weapons nonproliferation objectives to minimize the civil use of HEU worldwide.

Reprocessing of spent nuclear fuel is a well-established technology which is based on the same principles as chemical separation in the United States as discussed in Section 2.6.5.2, and an international commercial market has developed with a total annual capacity of several thousand MTHM (BNFL, 1994; Cogema, 1994). Large portions of this capacity are oriented toward commercial spent nuclear fuel. While these facilities are technically capable of reprocessing foreign research reactor spent nuclear fuel with relatively minor modifications [e.g., blending of the foreign research reactor spent nuclear fuel in the dissolver(s) with depleted uranium], for contractual, economic, and schedule considerations, these commercial spent nuclear fuel reprocessing facilities are less inclined to consider the potential foreign research reactor spent nuclear fuel market.

Several large (hundreds of MTHM per year) spent nuclear fuel reprocessing facilities exist for LEU spent nuclear fuel, and the annual capacity of existing facilities for research and HEU spent nuclear fuel is in the tens of MTHM range. The British facilities at Dounreay are currently capable of reprocessing foreign research reactor spent nuclear fuel. The French facilities at Marcoule are planning reprocessing of French research reactor spent nuclear fuel in the near future, and the Dounreay facility is pursuing additional contracts with foreign research reactor operators for reprocessing their spent nuclear fuel. The estimated schedule of foreign research reactor spent nuclear fuel shipments corresponds to a maximum of 2 MTHM per year, which could be accommodated by the existing reprocessing capacity for these fuels. Significantly, the most likely candidate facilities for reprocessing foreign research reactor spent nuclear fuel are located in Europe and are operated under Euratom and International Atomic Energy Agency safeguards. These facilities also offer full spent nuclear fuel management capabilities, including spent nuclear fuel storage prior to reprocessing, solidification of wastes, product and waste transportation, and assay adjustments (i.e., blending of HEU to LEU). Arrangements for shipment and disposition of the processing wastes would have to be implemented. Most processing contracts with European facilities require return of the wastes to the generator (in this case, the foreign research reactor operators) of the spent nuclear fuel.

DOE has considered the possibility of accepting, in the United States, the vitrified waste from the reprocessing of foreign research reactor spent nuclear fuel overseas. The environmental impacts from the receipt and storage of the waste in the United States are presented in Section 4.4.2.

2.4 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

In implementing the proposed action, DOE and the Department of State could combine implementation elements from the management alternatives analyzed in Sections 2.2 and 2.3. For example, DOE and the Department of State could consider partial storage or reprocessing overseas and partial storage or chemical separation in the United States. The impacts to the U.S. environment from hybrid alternatives would be bounded by the analysis presented in this EIS for each of the implementation alternatives for Management Alternative 1, because for each implementation alternative, the analysis considers the maximum amount of spent nuclear fuel that could be accepted, stored, and/or chemically separated in the United States.

For the purpose of illustration, DOE and the Department of State have considered an example of a hybrid alternative which is a combination of implementation elements of Management Alternatives 1 and 2. This hybrid alternative is described below.

Under this hybrid alternative, DOE and the Department of State would provide nontechnical (financial and/or logistical) assistance to foreign research reactor operators and reprocessors to facilitate the reprocessing of any foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2, for foreign research reactors in countries that could accept back the reprocessed waste; and DOE and the Department of State would accept and manage the rest of foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1.

In order to comply with U.S. nuclear weapons nonproliferation policy, bilateral agreements would need to be established with the foreign governments involved to ensure that the conditions discussed in Section 2.3 for overseas reprocessing (Management Alternative 2, Implementation Alternative 1b) would be met before DOE and the Department of State would consider implementation of this hybrid alternative.

Based on the current capabilities of the reprocessors overseas, the spent nuclear fuel to be considered for reprocessing would be aluminum-based. TRIGA spent nuclear fuel could also be considered if such capability is developed; however, for the purposes of the analysis, TRIGA spent nuclear fuel would be assumed to be accepted in the United States for storage. Table 2-3 lists the countries that may be able to accept the reprocessing waste and the amount of spent nuclear fuel to be considered for reprocessing overseas. Table 2-4 shows the amount of spent nuclear fuel that would be accepted in the United States.

Under this hybrid alternative, the aluminum-based foreign research reactor spent nuclear fuel to be accepted in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6, which is discussed in Section 2.2.2.6. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory, where it would be stored at existing storage facilities until ultimate disposition. The distribution of the spent nuclear fuel considered in this hybrid alternative is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

All the other components of this hybrid alternative are the same as the basic implementation of Management Alternative 1, specifically:

- a policy duration of 10 years with a period of acceptance of spent nuclear fuel in the United States of 13 years;

Table 2-3 Spent Nuclear Fuel Considered for Reprocessing Overseas (Hybrid Alternative Example)

Country	*Number of Elements*	*Mass (MTHM)*[a]
Belgium	1,766	0.730
France	1,962	3.442
Germany	1,504	0.909
Italy	150	0.043
Spain	40	0.016
Switzerland	159	0.128
United Kingdom	12	0.004
Total	**5,593**	**5.272**

[a] *To derive mass in kilograms, multiply by 1,000.*

Table 2-4 Amount and Distribution of Foreign Research Reactor Spent Nuclear Fuel to be Accepted in the United States (Hybrid Alternative Example)

Type	*Number of Elements*	*Mass (MTHM)*[a]	*Number of Shipments*
Aluminum-Based	12,210	12.912	406
Eastern[b]	7,593	8.647	275
Western[b]	4,617	4.263	131
TRIGA	4,940	1.033	162
Eastern[b]	3,245	0.528	107
Western[b]	1,695	0.505	55
Total	**17,150**	**13.945**	**568**

[a] *To derive mass in kilograms, multiply by 1,000.*

[b] *Refers to the location of the likely port(s) of entry to the United States.*

- a financing arrangement by which the United States would bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel received from developing countries, but would charge developed countries (if any) a competitive fee;
- taking title to the foreign research reactor spent nuclear fuel at the U.S. territorial waters limit or continental U.S. borders for shipments from Canada;
- marine transport of the foreign research reactor spent nuclear fuel by chartered and/or regularly scheduled commercial ships;
- ports of entry that qualify on the bases of criteria discussed in this EIS; and
- ground transport from ports of entry to the Savannah River Site and Idaho National Engineering Laboratory by truck, rail, or barge, or a combination of these modes.

The impacts to the U.S. environment from this hybrid alternative would be bounded by the Savannah River Site portion of Implementation Alternative 6 (Near Term Chemical Separation in the United States at the Savannah River Site), which considers the acceptance of approximately 22,700 elements of foreign research reactor spent nuclear fuel in the United States versus approximately 17,100 (13.9 MTHM) elements in this hybrid alternative; the chemical separation of approximately 17,800 aluminum-based elements versus approximately 12,200 aluminum-based elements in this hybrid alternative; and the storage of approximately the same number of TRIGA elements as under the basic implementation.

The environmental impacts and policy considerations of the hybrid alternative are discussed in Section 4.5.

2.5 No Action Alternative

In the No Action Alternative, the United States would neither manage foreign research reactor spent nuclear fuel containing uranium enriched in the United States, nor provide technical assistance or financial incentives for overseas storage or reprocessing. In this case, no foreign research reactor spent nuclear fuel shipments to the United States and no assistance to foreign countries for managing foreign research reactor spent nuclear fuel overseas would take place. The No Action Alternative would have environmental impacts outside the United States which are not in the scope of this EIS. Policy considerations are discussed in Section 4.6.

2.6 Characteristics of the Components of the Basic Implementation

This section summarizes information on the selection process for some of the components of the basic implementation, as well as characteristics, assumptions and physical parameters used in the environmental impact analysis. This section provides characteristics of the spent nuclear fuel to be received, the types of transportation casks considered, the marine ports considered and method of their identification, the ground transportation routes considered, method of identification, typical characteristics of the wet and dry storage technologies, descriptions of the designs of facilities for both dry and wet storage technologies, descriptions of chemical separation facilities, and details on the site-specific options in managing the foreign research reactor spent nuclear fuel.

2.6.1 Characteristics and Types of Foreign Research Reactor Spent Nuclear Fuel

Spent nuclear fuel is fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated. Fuel in a reactor consists of fuel elements that come in many configurations; but generally consist of the fuel matrix, cladding and structural parts. The matrix, which contains the fissionable material is typically in the form of plates or cylindrical pellets. The cladding (typically zirconium, aluminum, or stainless steel) surrounds the fuel, confining and protecting it. Structural parts (generally nickel alloys, stainless steel, zirconium, or aluminum) hold the fuel elements in the proper configuration in the reactor core.

Spent nuclear fuel is radioactive because of the presence of the radioactive isotopes, which are products of the fission process. The radiation of most concern from spent nuclear fuel is gamma rays. Although the radiation levels can be very high, the gamma ray intensities are readily reduced by shielding the spent nuclear fuel elements with such materials as steel, lead, concrete, and water during the various management phases of handling, transporting, or storing the spent nuclear fuel elements.

An issue associated with the management of spent nuclear fuel containing significant amounts of fissionable material is the potential for a self-sustaining nuclear fission process called criticality. Prevention of criticality conditions enters in the design of the spent nuclear fuel transportation casks, the spent nuclear fuel storage and processing facilities, and the spent nuclear fuel packaging for ultimate disposition. In general, criticality prevention is accomplished by either controlling the amount of fissionable material present within a certain volume (dilution or spatial separation techniques) or by introducing neutron absorbing materials that reduce the number of neutrons available to the fission process (poisoning technique). The criticality issue has been addressed in all implementation alternatives considered for the management of the foreign research reactor spent nuclear fuel.

Two types of foreign research reactor spent nuclear fuel are covered under the proposed action. They are aluminum-based fuel and TRIGA reactor type fuel. The aluminum-based fuel refers to fuels that consist of an alloy of uranium and aluminum, or a dispersion of uranium-bearing compound in aluminum, both clad in aluminum. The enrichment of uranium can be either HEU or LEU. Details on the physical and nuclear characteristics of the aluminum-based foreign research reactor spent nuclear fuel can be found in Appendix B. The aluminum-based fuels are used in various reactor types in different forms and geometries. The spent nuclear fuel element geometries are either cylindrical, boxed type, annular with hundreds of involute plates, or pin cluster. The aluminum-based fuel forms are either plates, tubes, rods, or pins. The ^{235}U[5] content of a fuel element prior to irradiation in a reactor (i.e., fresh fuel element) can vary from about 3 g (.11 oz) to about 8,500 g (300 oz). The length of an individual element can vary from 22 cm (8.7 in) to about 300 cm (118 in).

The TRIGA reactor fuel uses uranium-zirconium hydride (U-Zr-H_x) fuel material in which the hydrogen moderator is homogeneously contained within the fuel material. The initial ^{235}U content of each rod varies between 38 g (1.3 oz) and 133 g (4.7 oz). The overall length of a TRIGA fuel rod is approximately 76 cm (30 in), and the weight is between approximately 1 kg (2.2 lb) and about 4 kg (8.8 lb). Details on the physical and nuclear characteristics of the TRIGA foreign research reactor spent nuclear fuel can be found in Appendix B.

In contrast, a typical nuclear power reactor fuel (e.g., pressurized water reactor fuel) is three to five percent enriched uranium-oxide. The fuel form is ceramic pellets combined into rods, and the cladding is zircaloy or stainless steel. A typical pressurized water reactor fuel assembly weighs 682 kg (1,500 lb) and has a length of 389 cm (13 ft). Figure 2-5 graphically depicts the differences in size of a typical pressurized water reactor assembly, a typical aluminum-based fuel element, and a TRIGA fuel element. Additional detailed information on the aluminum-based and TRIGA fuels are provided in Appendix B of this EIS.

In addition to aluminum-based and TRIGA-type spent nuclear fuel, target material containing HEU is considered for management under Implementation Alternative 1, subalternative 1c (Section 2.2.2.1). Targets are irradiated in a research reactor to produce molybdenum-99, a medical isotope. Molybdenum production peaks at a low burnup, about three percent. Once the target is removed from the reactor, the fuel is dissolved in acid, and molybdenum-99 is separated from the solution. The residual material after removal of molybdenum-99 is called target material, and is currently kept in solution form. The target material considered for management would be put in U_3O_8 or UO_2 form and canned for transport to the United States. It is expected that the target material would contain about 0.6 MTHM (the uranium content of 620 typical Material Test Reactor [MTR] elements) and a volume of 6.5 m^3 (229.5 ft^3). This material could be brought to the United States in cans having a cavity of 6.4 cm (2.5 in) in diameter and 28 cm (11 in) long, and containing between 40 g to 200 g (1.41 oz to 7 oz) of ^{235}U each.

2.6.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages called transportation casks, or just casks. A typical cask for the transportation of foreign research reactor spent nuclear fuel elements is shown in Figure 2-6. Detailed descriptions of typical casks are provided in Appendix B (Section B.2).

5 During the enrichment process, the amount of fissionable Uranium-235 (^{235}U) is increased. Uranium increased to less than 20 percent ^{235}U is called LEU. Uranium enriched to 20 percent or greater ^{235}U is called HEU.

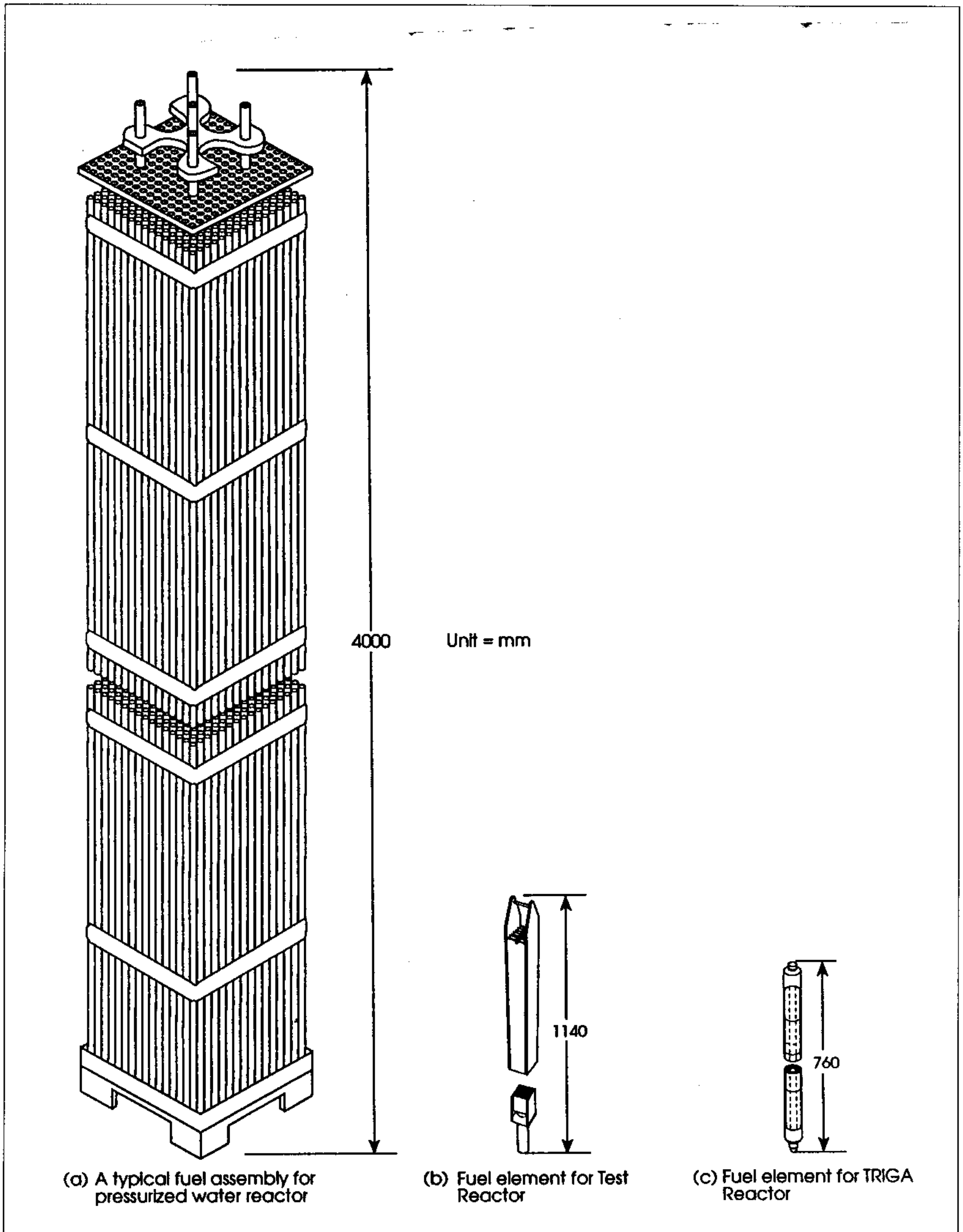


Figure 2-5 Typical Spent Nuclear Fuel Elements

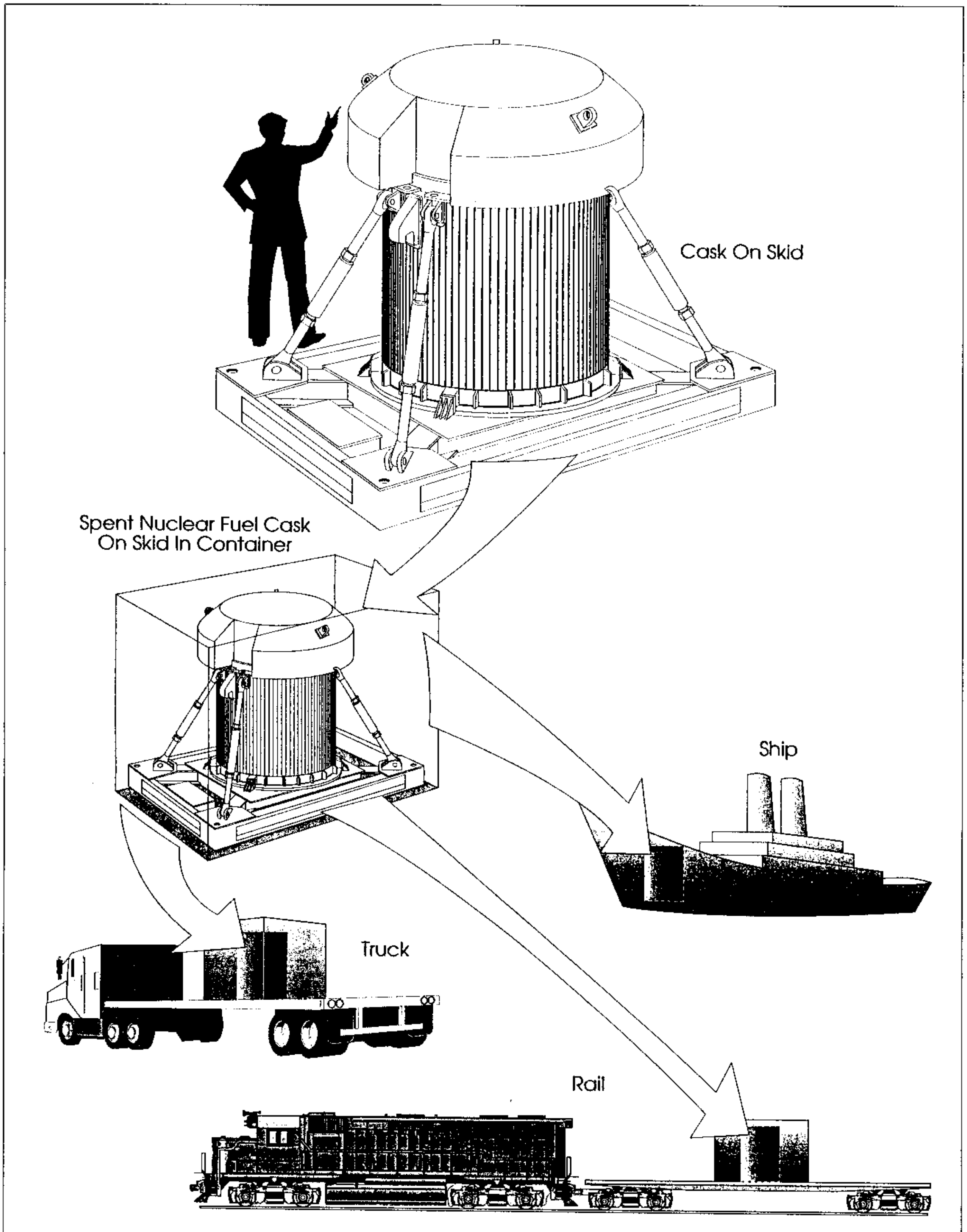


Figure 2-6 Typical Foreign Research Reactor Spent Nuclear Fuel Transportation Cask

Table 2-5 Representative Transportation Casks for Foreign Research Reactor Spent Nuclear Fuel

Transportation Cask	*Country of Origin*	*Certificate of Compliance*	*Estimated Capacity Number of Elements*
Marine Transport			
LHRL-120	Australia	USA/0389/B(U)F	114
GNS-11	Germany	USA/0381/B(U)F	21-33
TN-1	Germany	USA/0316/B(U)F	126
IU-04	France	USA/0100/B(U)F	36-40
TN-7 (TN-7/2)	Germany	USA/0130/B(U)F	60-64
NAC-LWT	United States	USA/9225/B(U)F	48-64
UNIFETCH	United Kingdom	GB/1113/B(M)F	24-40
GOSLAR	Germany	USA/0094/B(M)F	13
Ground Transport (Between Sites)			
NL1-10/24	United States	USA/9023/B( )F	120-160
IF-300	United States	USA/9001/B( )F	84-112
BMI-1	United States	USA/5957/B(U)F	24
GE-2000	United States	USA/9228/B(U)F	24
TN-8	Germany	USA/9015/B()F	36-48
NLI-1/2	United States	USA/9010/B( )F	48-64
NAC-LWT	United States	USA/9225/B(U)F	48-64

A full cask can carry from 13 to 120 spent nuclear fuel elements from foreign research reactors, depending on fuel element design, size, and cask capacity. The casks are certified as "Type B" under regulations. To receive this certification, a cask must successfully pass tests simulating severe accident conditions. The tests include being dropped onto an unyielding surface, dropped onto a steel post, subjected to extremely high temperatures of 802°C (1,475°F) for 30 minutes, and submersion in water for 8 hours.

As discussed in Section 2.7.2, "Type B" casks have been used for years to transport spent nuclear fuel elements within the United States and around the world (DOE, 1994d). To date, no spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive contents, even in actual highway accidents (NRC, 1993).

The casks are designed to provide shielding from radiation. However, a low radiation field is present outside the cask — usually less than one mrem per hour at 1 m (3.3 ft) away from the cask.

Table 2-5 identifies typical transportation casks that could potentially be used for transporting foreign research reactor spent nuclear fuel from the foreign research reactor sites to the candidate United States ports of entry and to the potential management sites. A majority of these have already been used by DOE for transporting foreign research reactor spent nuclear fuel.

As explained in Section 2.6.4, the inability of certain management sites to accept foreign research reactor spent nuclear fuel at the beginning of the implementation period could necessitate temporary (as long as 10 years from the start of the policy) management of foreign research reactor spent nuclear fuel at an available site and the eventual transport of this foreign research reactor spent nuclear fuel to another site. At the time of the intersite transport, the foreign research reactor spent nuclear fuel elements would contain less radioactivity and less heat because of the decay process. The transportation, therefore, could be carried out in casks with larger capacity than those considered for marine transport. Such casks, currently licensed only for the transportation of commercial spent nuclear fuel in the United States, would need to be certified for foreign research reactor spent nuclear fuel. The size of these casks would allow the transport of up to 4 (truck-size casks) or up to 10 (rail-size casks) times as many elements in a single shipment as those considered for the marine transport casks. The bottom part of Table 2-5 identifies typical casks for intersite transportation. Description and design information is included in Appendix B (Section B.2).

2.6.3 Marine Transport and Ports

This section describes the potential activities related to foreign research reactor spent nuclear fuel marine port identification and marine transport activities.

2.6.3.1 Marine Port Identification

In this EIS, port screening and selection were performed to identify candidate ports of entry for the foreign research reactor spent nuclear fuel. The criteria used in this process were based on several sources, including:

- A DOE-sponsored workshop on port selection criteria for spent nuclear fuel held at the U.S. Merchant Marine Academy at Kings Point, NY, on November 15-16, 1993 (USMMA, 1994).
- Public input to the scoping meetings for this EIS, as summarized in the DOE Implementation Plan (DOE, 1994h).
- Factors identified in Section 3151 of the National Defense Authorization Act for Fiscal Year 1994.

These sources are described in more detail in Appendix D. After consulting the above-mentioned sources, a list of criteria for ports eligible to receive spent nuclear fuel was developed. These criteria are:

- Appropriate port experience - port terminal(s) and operators should routinely handle containerized dry cargoes that require the same type of handling as containerized spent nuclear fuel, or will have the capability to handle these cargo types during the proposed management policy period;
- Port transit - the port should be within reasonable distance from the open sea, with a good ship channel;
- Appropriate port facilities - the port should have adequate crane(s), piers, and depth of water alongside the pier;
- Ready intermodal access - the port should have ready access for intermodal transport; and
- Low human population - the human population of the ports and along transportation routes to potential management sites should be low to the extent feasible and maximum extent practicable.

These criteria, taken collectively, provided DOE and the Department of State with the basis for identifying and analyzing potential ports of entry. Additionally, port identification was expanded (i.e., the NWS Charleston was added to the Port of Charleston) as a result of public comment on the Draft EIS.

2.6.3.2 Marine Transport and Port Activities

2.6.3.2.1 Marine Transport

DOE and the Department of State estimate that approximately 721 cask loads of foreign research reactor spent nuclear fuel would be sent to the United States by ship over the 13-year acceptance period under the basic implementation of Management Alternative 1. The International Maritime Organization currently limits the typical commercial cargo ship (Class INF-2) to a maximum of 200 petrabecquerels of radioactivity (IMO, 1993), which equates to approximately 5.4 million Ci. A typical cask of foreign research reactor spent nuclear fuel is predicted to contain 1 million Ci (see Appendix C). Therefore, a shipment in a commercial cargo ship could contain several casks.

Two types of analysis were conducted to evaluate the impacts of the marine transport of foreign research reactor spent nuclear fuel: first, assuming there are no accidents (incident-free); second, assuming various accidents occur. The incident-free analyses were conducted for ships' crews and port workers, assuming ships carrying two and eight casks of foreign research reactor spent nuclear fuel. Accident analyses were conducted for accidents in port and for accidents in coastal waters and the open ocean. The number of shipments is a parameter of primary importance in the incident-free analysis as well as the accident analysis in port, coastal waters, and open ocean. As noted above, 721 shipments were considered for the basic implementation of Management Alternative 1. In implementing the proposed policy, DOE would attempt to minimize the number of shipments by maximizing the number of casks that would be carried in a single shipment. However, for the purpose of assessing the environmental impacts, a single-cask per shipment assumption is made for the purpose of conservatism. The number of shipments for the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3 discussed in Section 2.4 are roughly proportional to the amount of foreign research reactor spent nuclear fuel to be accepted in the United States under each alternative. The exact number of shipments assumed in the analysis is provided in Appendices C and D, Section C.4 and D.4, respectively. The results of both the incident-free and the accident analyses are presented in Chapter 4, with details in Appendices C and D.

There are four types of ships that could be used to transport foreign research reactor spent nuclear fuel casks. These are:

Container vessels: These are typically large ships specifically intended for the transport of containerized cargo. Some modern container ships can transport up to about 5,000 containers, although a more typical capacity is in the range of 800 to 1,000 containers. A principal advantage of container ships, because of standardization of containers, is that the vessel can be rapidly loaded or off-loaded at those ports equipped with container gantry cranes. Containers can be removed from, or placed on, the vessel at an average rate of about 45 containers per hour. At well-equipped container ports, two cranes are used to move containers.

Roll-on/roll-off ships: These ships are vehicle carriers used for the ocean transport of cars and trucks. The ships are loaded and unloaded using a ramp between the vessel and dock. Typically, the vessel carries its own ramp, which is deployed by an on-board crane, hydraulic cylinders, or chain drives. The ramp may extend from the stern of the vessel or from a hatch in the side of the vessel hull. At docks intended for roll-on/roll-off service, additional ramps may be deployed from the dock to expedite loading or unloading. This type of ship could carry foreign research reactor spent nuclear fuel casks secured on trailers.

General/cargo (breakbulk) ships: General cargo vessels are smaller ships that typically call on less well-developed or equipped ports. They have on-board jib or boom type cranes that can be used to load or unload the ship if dockside crane service is not available. As the name implies, these vessels are intended to accommodate a wide variety of cargoes that may have any reasonable configuration. Since the advent of the widespread use of containers, most of these ships are equipped with lock fixtures to secure containers during transport. If necessary, containers can be lifted on and off these ships by using four-legged slings between the corners of the container and hook of the crane.

Purpose-built ships: For the purposes of this EIS, the ships discussed here are specifically designed to transport spent nuclear fuel casks. These ships are not used for the transport of any other cargo, and they operate as chartered vessels. Casks are loaded directly into the holds of the ship because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the ship has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle only one cask type, however, other casks may be used with minor modifications. For the relatively efficient transport of spent nuclear fuel, the casks are large. These type vessels are intended for the transport of commercial power nuclear reactor fuel, and they generally operate between nuclear installations (power plants and spent nuclear fuel end-use facilities) having dedicated docks. Commercial docks are not normally used, but could be. These vessels have double bottoms and hulls and collision damage-resisting structures within the hull. The vessel crew is trained in the handling of the cargo and in emergency response like most other commercial vessels.

The potential exists that spent nuclear fuel would be accepted from all 41 countries that have expressed interest in this program. Ships carrying the foreign research reactor spent nuclear fuel would follow normal shipping routes from a convenient port in or near the country of origin, and would go to a U.S. port that is consistent with the port identification, evaluation, and selection process as described in Appendix D.

Regularly scheduled commercial service cargo ships could be used to ship foreign research reactor spent nuclear fuel. Some, if not most, of the regularly scheduled commercial ships might initially call at a port other than the port of destination of the foreign research reactor spent nuclear fuel, and may make additional stops. Therefore, marine transport may involve entry into and departure from intermediate ports and shipping in coastal waters. Typically, ships spend 1 day in each port of call and 1 or 2 days passing between ports.

Risks to the ships carrying foreign research reactor spent nuclear fuel and to the spent nuclear fuel itself can arise from natural sources, such as storms at sea, and from other events, such as collisions with other ships and marine obstacles, as well as from fires. Modern technology and good communications help minimize these risks by keeping ships informed of severe weather and other shipping and marine obstacles. Risk to the cargo is further reduced through proper stowing and securing, and through daily cargo inspections while at sea to ensure that the cargo remains secured.

Regardless of the technology and practices mentioned above, accidents involving ships carrying foreign research reactor spent nuclear fuel would be possible. Consequences of accidents at sea have been evaluated and are discussed in Chapter 4, and described in more detail in Appendix C.

The presence of a cask containing foreign research reactor spent nuclear fuel onboard could result in a radiation dose to some of the ship's crew due to radiation that emanates from the cask. Most of the ship's crew would be relatively far away from the cargo (and the cask), and therefore, would receive essentially no radiation dose. However, the daily inspection of the cargo would bring an inspector in close proximity to the cask containing the foreign research reactor spent nuclear fuel for a short period of time. Effects on inspectors and other incident-free impacts were evaluated, and are described in Chapter 4 and detailed in Appendix C.

Both commercial and military ports were evaluated for potential use as ports of entry for the foreign research reactor spent nuclear fuel. DOE determined that the security provisions specified by 10 CFR 73, which are required for all spent fuel shipments, could be implemented at either commercial or military ports. Any additional security that might be available at a military port would not be required for foreign research reactor spent nuclear fuel shipments.

2.6.3.2.2 Port Activities

Entry into a port by commercial vessels is accomplished under the control of the port authority. The port authority is responsible for the area from the sea buoy to the dock, except where the approaches are long, such as the case with the Chesapeake Bay. Normally, each ship is required to have a pilot familiar with local conditions to direct it while underway in the harbor or channel and during both entry and exit approaches. The pilot's job is to ensure that the ship follows the marked channel and arrives safely at its dock or other assigned location. In the event of bad weather or low visibility, radar and other instruments are available on all ships that would be considered for carrying foreign research reactor spent nuclear fuel.

In most cases, the ship moves directly to its assigned dock. However, if the assigned dock is still occupied or is not immediately available for other reasons, the ship may anchor in the harbor or its approaches for a period of time prior to docking. Most ocean-going vessels are not highly maneuverable in confined spaces, so docking is normally accomplished with the help of one or more tugs.

At the ship's first port of call in the United States, the U.S. Coast Guard and other authorities would inspect the ship, its cargo, and documentation. In the case of radioactive cargoes, the NRC may inspect the container with the radioactive material and its documentation. State and local officials could also perform inspections of documents and cargo. At ports of call after the initial port(s) of entry, additional inspections may be performed by Federal, State, or local officials.

Except for roll-on/roll-off ships, all cargo ships that would potentially carry foreign research reactor spent nuclear fuel are unloaded with cranes. Unloading of a foreign research reactor spent nuclear fuel cask, whether in a container or not, involves connecting a lifting fixture to the container or to the cask pallet, lifting the container or cask, and placing it dockside, either on an intermediate vehicle or directly on the primary mode of land transportation. Typically, container ships can be unloaded at the same rate as they are loaded (approximately 45 containers per hr), while unloading a cask on a handling platform in a breakbulk ship would require more time.

All shipments of foreign research reactor spent nuclear fuel would be anticipated well in advance, so the container housing the foreign research reactor spent nuclear fuel cask would normally be loaded immediately on the ground transportation to be used to carry it out of the port. Should there be a delay, the container may be temporarily stored at the port for up to 24 hours. Port security at any of the ports selected for analysis is adequate to protect the container in the event of this unexpected delay.

In spite of all of the precautions taken, accidents in the port would be possible. In fact, most ship accidents occur in or around ports. DOE has had no radioactivity released in the past due to port accidents; however, a range of accidents, both at the dock and in the port or its approaches, has been evaluated. See Chapter 4 for a discussion of the results of these analyses.

All ports considered for receiving foreign research reactor spent nuclear fuel would have emergency plans for responding to an accident in the port.

2.6.4 Ground Transport Route Options and Route Identification Process

2.6.4.1 Ground Transport Route Options

Route options for the potential ground transportation of foreign research reactor spent nuclear fuel depend on the marine ports considered, the management sites and the various ways that foreign research reactor spent nuclear fuel would be distributed among the potential management sites according to the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, routes and the amount of fuel to be shipped would be established based on one of the following spent nuclear fuel distributions:

- an even distribution of foreign research reactor spent nuclear fuel between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and the 1992/1993 Planning Basis alternatives;
- a distribution that sends TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory and aluminum-based spent nuclear fuel to the Savannah River Site under the Regionalization by Fuel Type alternative;
- a distribution that sends the spent nuclear fuel entering the United States from the Eastern ports to the Savannah River Site or the Oak Ridge Reservation and the spent nuclear fuel entering the United States from the Western ports to the Idaho National Engineering Laboratory, the Nevada Test Site, or the Hanford Site under the Regionalization by Geography alternative; or
- a distribution that sends all foreign research reactor spent nuclear fuel to one of the five potential management sites under the Centralization alternative.

For the purposes of this EIS, the distribution of foreign research reactor spent nuclear fuel between sites under Regionalization by Geography and by Fuel Type has been analyzed in detail. The more detailed planning performed in preparation for the analyses of the various alternatives considered in this EIS did not reveal any physical situation in which an even distribution of the spent nuclear fuel between two sites was advantageous. Furthermore, the impacts of activities associated with the even distribution at either site would be bounded by and equal to roughly 50 percent of the impacts of the centralization of all foreign research reactor spent nuclear fuel management activities at that site. As a result, this alternative is not analyzed in detail in this EIS.

An additional factor which would affect the route options for ground transportation is the inability of certain potential spent nuclear fuel management sites to implement the foreign research reactor spent nuclear fuel management policy immediately. Of the five sites, only two (the Savannah River Site and the Idaho National Engineering Laboratory) would be immediately available in late 1995. The other three could become available at a later date when appropriate facilities for accepting and managing foreign research reactor spent nuclear fuel become available. This constraint affects the ground transportation route options in the case that a site, other than the Savannah River Site or the Idaho National Engineering Laboratory, is considered and for DOE's spent nuclear fuel management under either the Regionalization by Geography or the Centralization alternative. If the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site is one of the management sites, the foreign research reactor spent nuclear fuel would have to be shipped first to one of the available management sites (the Savannah River Site and/or the Idaho National Engineering Laboratory) and later, when appropriate facilities are completed, to either the Nevada Test Site, the Oak Ridge Reservation, or the Hanford Site.

Certain assumptions are required in order to simply and consistently describe the manner in which foreign research reactor spent nuclear fuel would be transported to the management sites. The shipments, which were identified earlier in Tables 2-1 and 2-2, were divided into east coast and west coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East, and parts of Central and South America were designated as east coast shipments. All others were designated as west coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the point of origin in Canada. Under these assumptions, for the basic implementation of Management Alternative 1, the Eastern points of entry would receive 651 cask shipments (535 from ports, 116 from Canada) and the Western ports of entry would receive 186 cask shipments (all from ports).

No intersite shipments would be necessary under the Programmatic SNF&INEL Final EIS alternatives (DOE, 1995c) that use the Savannah River Site and/or the Idaho National Engineering Laboratory for managing the foreign research reactor spent nuclear fuel. The estimated number of shipments for the basic implementation of Management Alternative 1 in these cases would be as follows:

- Decentralization, 1992/1993 Planning Basis, or Regionalization by Geography to the Savannah River Site and the Idaho National Engineering Laboratory - the Savannah River Site would receive 651 casks from the east coast and the Idaho National Engineering Laboratory would receive 186 casks from the west coast.

- Regionalization by Fuel Type - the Savannah River Site would receive 675 casks of aluminum-based fuel; 544 from the east coast and 131 from the west coast. The Idaho National Engineering Laboratory would receive 162 casks of TRIGA-type fuel; 107 from the east and 55 from the west.

- Centralization to the Idaho National Engineering Laboratory or Centralization to the Savannah River Site - the site would receive 837 casks; 651 from the east coast and 186 from the west coast.

A two-phased program would be required if a site other than the Idaho National Engineering Laboratory or the Savannah River Site is considered as a central or regional site. Phase 1 is defined as the period from the beginning of the policy (late 1995) until the Phase 2 site (the Hanford Site, the Nevada Test Site and/or the Oak Ridge Reservation) would be ready to receive fuel, which is estimated to be 10 years for new construction; less time would be required for refurbishment of an existing facility. During Phase 1, DOE would manage the fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory. During Phase 2, DOE would ship any fuel that is being managed during Phase 1 at a non-Phase 2 site to a Phase 2 site, and manage the fuel at that site until a repository becomes available. The phases are defined to help describe the implementation of the foreign research reactor spent nuclear fuel management policy and to analyze the transportation impacts of the implementation of the policy.

If the Hanford Site, the Nevada Test Site, and/or the Oak Ridge Reservation were selected under the Programmatic SNF&INEL EIS, DOE and the Department of State would select from the following four strategies for managing fuel at the Savannah River Site and/or the Idaho National Engineering Laboratory during Phase 1. DOE could: (1) divide the fuel by geography, (2) divide the fuel by type (aluminum-based and TRIGA), (3) ship all fuel to the Savannah River Site, or (4) ship all fuel to the Idaho National

Engineering Laboratory. Therefore, in Phase 2, the Hanford Site and the Nevada Test Site could receive all foreign research reactor spent nuclear fuel, or TRIGA or Western spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1. Similarly, the Oak Ridge Reservation could eventually receive all foreign research reactor spent nuclear fuel or the aluminum-based or Eastern spent nuclear fuel managed at the Savannah River Site during Phase 1.

An assumption on the rate at which spent nuclear fuel arrives is necessary to estimate the number of shipments that would arrive during Phases 1 and 2. The demand to ship fuel by the foreign research reactor operators would be highest at the beginning and the end of the proposed policy period. However, the limited availability of casks would compel DOE to receive fuel at a steady rate. DOE could control the rate at which fuel is delivered by managing the contracts with shippers. Therefore, for the purposes of this analysis, it would be reasonable to assume that the 837 casks would arrive at a uniform rate during a 13-year period. Based on this rate of about 65 casks per year, it is estimated that 644 casks would be received during Phase 1 (approximately 10 years), and 193 casks would be received during Phase 2.

The projected number of shipments for the two-phased regionalization and centralization approaches are shown in Tables 2-6 and 2-7. The projections are based on the types and locations of spent nuclear fuel described in Appendix B, and the strategies and arrival rate assumptions described above. Each projection is described in more detail and shown on a map in Appendix E.

As noted earlier, the impact analysis from transportation depends on the location of entry (Eastern or Western ports) and number of shipments that would reach the United States. The discussion above pertains to the basic implementation of Management Alternative 1. In considering the implementation alternatives discussed in Sections 2.2.2.1 through 2.2.2.7 and Management Alternative 3, discussed in Section 2.4, both the number of shipments and locations of entry would vary with each alternative. The detailed distribution and number of shipments assumed to set up the ground transportation routes for each alternative are provided in Appendix E, Section E.8.

2.6.4.2 Route Analysis

Foreign research reactor spent nuclear fuel shipments would have to comply with both NRC and Department of Transportation regulatory requirements. The highway routing of spent nuclear fuel is systematically determined in accordance with Department of Transportation regulations [49 CFR 171-179 and 49 CFR 397].

The Department of Transportation routing regulations require that these shipments be transported over a preferred highway network including:

- Interstate highways;
- An interstate system bypass or beltway around a city; or
- State-designated preferred routes.

The selection of the preferred highway routes are consistent with the U.S. Department of Transportation's published guidelines (DOT, 1992).

In addition to defining routes, 49 CFR Part 397 contains the driver safety requirements for highway carriers of packages of radioactive material exceeding a quantity of material known as a "highway route-controlled quantity." All spent nuclear fuel shipments would be expected to exceed this quantity.

Table 2-6 Shipment Summary for Regionalization by Geography Alternatives

Spent Nuclear Fuel Site Option Western/Eastern	Phase 1 Approach	Phase 1 Port-to-Site Shipments	Site-to-Site Shipments[a]	Phase 2 or Port-to-Final Site Shipments	Total Number of Shipments[a]
INEL/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51	East to ORR: 150 West to INEL: 43	963/888
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52	East to ORR: 150 West to INEL: 43	967/889
	All to INEL	644	None	East to ORR: 150 West to INEL: 43	837
NTS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to NTS: 36/15	East to SRS: 150 West to NTS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to NTS: 31/13	East to SRS: 150 West to NTS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to NTS: 43	837
NTS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to NTS: 36/15	East to ORR:150 West to NTS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to NTS: 31/13	East to ORR: 150 West to NTS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to NTS: 43	998/902
	All to INEL	644	INEL to NTS: 161/65	East to ORR: 150 West to NTS: 43	998/902
HS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to HS: 36/15	East to SRS: 150 West to HS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to HS: 31/13	East to SRS: 150 West to HS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to HS: 43	837
HS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to HS: 36/15	East to ORR: 150 West to HS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to HS: 31/13	East to ORR: 150 West to HS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to HS: 43	998/902
	All to INEL	644	INEL to HS: 161/65	East to ORR: 150 West to HS: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments, assuming that the truck casks used for interstate shipments are capable of carrying 4 times as much fuel, and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor.*

Rail routing is not covered by specific Department of Transportation and NRC regulations. Therefore, carriers would generally select the most direct route, which would serve to reduce travel time and radiation exposure consistent with track class and other rail service requirements.

NRC regulations concerning physical security and notification are set forth in 10 CFR 71 and 10 CFR 73, respectively. Carriers are required to submit proposed routes for spent nuclear fuel shipments to NRC for approval, and NRC publishes a public information circular that lists routes that have been evaluated and approved for specific spent nuclear fuel shipments (NRC, 1993).

Table 2-7 Shipment Summary for Centralization Alternatives

Spent Nuclear Fuel Site Option	*Phase 1 Approach*	*Phase 1 Port-to-Site Shipments*	*Site-to-Site Shipments*[a]	*Phase 2 or Port-to-Final Site Shipments*	*Total Number of Shipments*[a]
SRS	N/A - Single phase			837	837
INEL	N/A - Single phase			837	837
HS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
ORR	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
NTS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

[a] *Truck/rail shipments assuming that the truck casks used for intersite shipments are capable of carrying 4 times as much fuel and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor due to consolidation.*

The HIGHWAY and INTERLINE computer codes are used to assist in route selection and estimations of exposed population (DOE, 1995c). The collective population risk, maximally exposed individual (MEI) risk, accident risk, accident consequence, and nonradiological risk assessments are performed using the RADTRAN and RISKIND computer codes established for shipment by both railroad and highway. Additional details of the treatment and analysis methodology used in the ground transportation assessment are given in Appendix E.

2.6.5 Activities and Alternatives at the Foreign Research Reactor Spent Nuclear Fuel Management Sites

The potential sites for receipt and management of foreign research reactor spent nuclear fuel are the same as those considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

Since foreign research reactor spent nuclear fuel is part of the overall DOE spent nuclear fuel management program, the potential site-specific options are consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River sites), Regionalization (distribution by fuel type and geography), and Centralization (all foreign research reactor spent nuclear fuel at the potential site).

As discussed earlier, the site-specific foreign research reactor spent nuclear fuel management options also depend on the availability of the management sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory will be available in late 1995. The other three could become available at a later date when construction or refurbishment of appropriate facilities is completed. This constraint has resulted in the two-phased approach considered in some cases (see discussion in Section 2.6.4.1). For the purpose of the site impact analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and management of foreign research reactor spent nuclear fuel is accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities are used (Phase 2). For the environmental impact analysis, the first is characterized by operational activities only, while the second involves impacts from construction and operation activities.

Section 2.6.5.1 provides an overview of the storage technologies and descriptions of the storage facilities considered under the implementation alternatives of Management Alternatives 1 and 3. Section 2.6.5.2 provides a description of chemical separation, which is considered as an implementation alternative to storage.

The site-specific options selected for impact analysis are described separately in the sections devoted for each site (Sections 2.6.5.3.1 through 2.6.5.3.5).

2.6.5.1 Storage Technologies

The purpose of a spent nuclear fuel management facility is to provide an environment for the storage of spent nuclear fuel that protects the public, onsite workers, and the environment. The principal hazard presented by spent nuclear fuel is its inventory of radioactive elements that are the products of the reactions in a nuclear reactor. In addition, the fissionable uranium and plutonium remaining in the spent nuclear fuel has the potential of sustaining a fission chain reaction, which would generate additional radiation and fission products.

The management facility is designed to prevent the stored spent nuclear fuel from achieving a fission reaction (termed "criticality") and to isolate the radioactive materials within the spent nuclear fuel from the public and workers. Criticality is prevented by such methods as:

- maintaining a minimum separation distance between adjacent spent nuclear fuel elements;
- limiting the concentration of fissionable materials in each spent nuclear fuel storage container;
- installing neutron-absorbing materials between spent nuclear fuel elements; and
- controlling the presence and/or concentration of other materials that would enhance the ability of the stored spent nuclear fuel to become critical.

Protection of the public and workers from the radioactive materials within each spent nuclear fuel element is achieved by:

- enclosing or encapsulating the spent nuclear fuel so that any accidental release of radioactive material is retained;
- maintaining a benign chemical and thermal environment around the spent nuclear fuel so that its structural integrity is preserved;
- providing adequate shielding of the radiation emanating from the spent nuclear fuel so that dose rates outside the facility are lowered; and
- utilizing security barriers to isolate spent nuclear fuel from workers and public.

The technology for safely storing spent nuclear fuel (as defined by the above criteria) has been in use, in one form or another, for over 40 years in the nuclear industry. Spent nuclear fuel storage is generally characterized as either wet or dry, denoting whether the spent nuclear fuel elements reside in a water-filled pool or a dry atmosphere. Details of the concepts are provided in Appendix F, Section F.1.

The wet pool type of spent nuclear fuel storage is used at almost every water-cooled nuclear reactor in the world. There are currently more than 600 operating water-cooled power and research nuclear reactors, each with an individual storage pool. The pool design uses common materials (water and concrete) for spent nuclear fuel shielding, heat removal, and the confinement of any radioactive material that might be released from the spent nuclear fuel. An additional benefit is the ability to visually inspect spent nuclear fuel, since the water purity and clarity are maintained at a high level. Spacing, fissionable material limits, and in some cases, the use of neutron-absorbing material prevent criticality in a wet storage environment. The pool is enclosed in a suitably qualified structure or building. Construction of a wet storage facility involves excavating earth, backfilling, pouring concrete, setting piping, erecting a building around the pool, and installing piping, electrical systems, and heating, ventilating, and air conditioning systems. In many ways, a spent nuclear fuel storage pool is like a swimming pool, except its depth is greater and its concrete walls and floors are much thicker to provide for structural integrity. Wet storage facility designs include sophisticated methods of leak detection. To negate corrosion, the pool water purity and quality are carefully maintained and controlled.

Dry storage technology involves the encapsulation of spent nuclear fuel in a steel cylinder that may be placed in a concrete or massive steel cask or structure. The spent nuclear fuel is stored in racks within the cylinder or suspended from plates placed at variable distances in the cylinder, in either air, or inert atmosphere. Foreign research reactor spent nuclear fuel elements with suspect cladding integrity would be placed in sealed cans before they are placed in the cylinder (canning). Casks or structure materials, usually some form of concrete, steel, iron, or lead provide shielding and heat removal. Spacing, fissile material limits, and neutron absorbing materials are used to prevent criticality. Different forms of dry fuel storage have been used for over 40 years in the nuclear industry. Several nuclear power plants in the United States have licensed, built, and operated dry storage facilities during the last 7 years. NRC has reviewed and approved several manufacturers' designs for dry fuel storage of commercial spent nuclear fuel. Canada has been storing spent commercial nuclear power plant fuel in dry storage casks since 1975. Australia has been successfully storing its research reactor spent nuclear fuel since 1963 (Silver, 1993) in dry environment, and Japan has had 12 years of experience with dry storage of research reactor spent nuclear fuel (Shirai et al., 1991). The Savannah River Site has an ongoing developmental program on dry storage technology which would be used to implement this worldwide experience in the United States, and to finalize design parameters for a foreign research reactor spent nuclear fuel dry storage facility.

Dry storage methods are not as efficient in removing heat from the spent nuclear fuel as wet storage pools. Thus, as explained in Appendix F, this EIS assumes that high decay heat foreign research reactor spent nuclear fuel would initially be placed in wet storage. This would allow sufficient time for the spent nuclear fuel radioactive decay heat to decrease and not be a deciding factor in sizing a dry storage facility.

Dry storage facility construction involves the preparation and pouring of concrete foundations upon which the concrete or metal cask or building is then erected. Metal casks would be built away from the DOE site in a factory, since they involve thick metal fabrication techniques not used at DOE facilities. Concrete casks or buildings are constructed at the site using the same general principles (e.g., forms, rebar) as in nonnuclear concrete construction. Qualified concrete foundation pads are also poured for support bases of the casks.

Whether wet or dry storage were used, the facility would be designed to withstand natural phenomena such as earthquakes, floods, tornadoes, hurricanes, high and low temperatures, and wind generated missiles (branches, poles, etc.). The design would also include provisions to preclude sabotage or terrorist acts. Security requirements for a dry storage facility after the spent nuclear fuel has decayed to low levels of radioactivity and is no longer self-protecting would be met by the establishment of a Perimeter Intrusion Detection and Alarm System zone, which is the standard procedure for DOE. Each design has specific provisions for periodic inspection or surveillance, and must meet the highest quality standards associated with all safety requirements specified for nuclear facilities.

The current alternative types of storage technology are discussed and evaluated in detail in Appendix F. The basic categories are: wet (pool), dry concrete vault or building, dry concrete horizontal cask/module, dry concrete vertical cask/silo, dry metal vertical cask, hot cells, multi-purpose casks, and dry inground vertical holes. There are significant differences between these technologies in terms of construction, operations and maintenance costs and various design details. However, these differences do not result in any important variations in environmental impacts and consequences. With the exception of multi-purpose casks (which are still under development), all of these technologies have proven records of successful safe operation while storing spent nuclear fuel. Appendix F provides detailed descriptions concerning generic dry and wet storage facilities for foreign research reactor spent nuclear fuel. Brief descriptions of both wet and dry storage facilities are provided in the following sections.

2.6.5.1.1 Description of Dry Storage Facilities

Spent Nuclear Fuel Storage Using a Modular Dry Vault:

An aboveground dry vault is a self-contained concrete structure that would allow for dry spent nuclear fuel handling and storage. This design represents an integrated spent nuclear fuel storage approach and would consist of four major components: a receiving/loading/inspection area, spent nuclear fuel storage canisters, a shielded canister handling machine, and a modular array for storing the spent nuclear fuel storage canisters. Figure 2-7 displays an illustration of a typical modular dry vault storage facility. The receiving area would use a wet pool for unloading the casks and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel elements with a heat load exceeding 40 Watts per element. The vault would consist of several modular units, and each unit could provide storage for hundreds of spent nuclear fuel assemblies. The vault itself would contain a charge/discharge bay with a spent nuclear fuel handling machine above a floor containing steel tubes that house the (removable) spent nuclear fuel canisters. The bay would be shielded from the stored spent nuclear fuel by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes would serve as secondary containment for the foreign research reactor spent nuclear fuel and would descend into an open storage

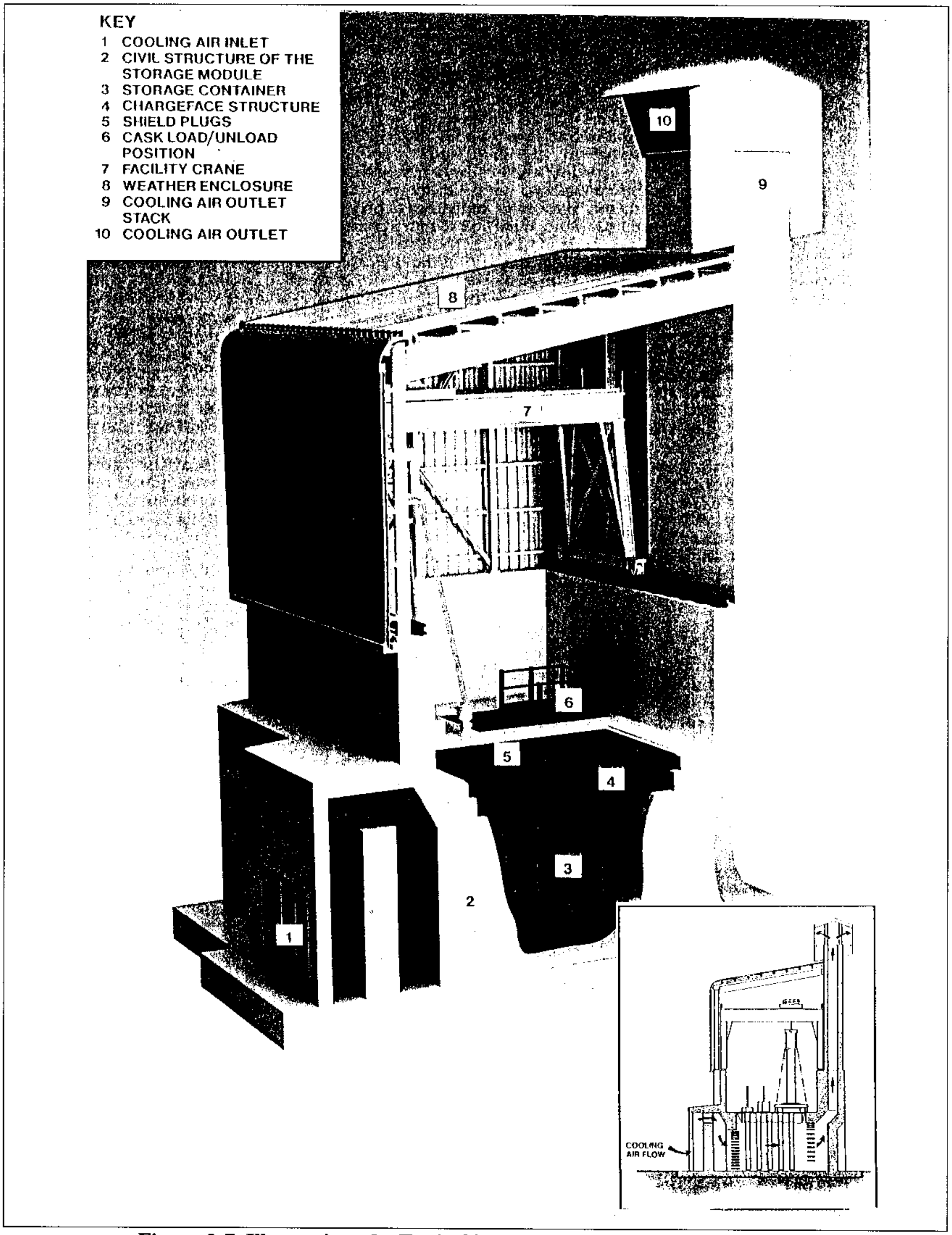


Figure 2-7 Illustration of a Typical Modular Dry Vault Storage Facility

Table 2-8 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility: size (area) concrete steel	 5,000 m^2 (54,000 ft^2) 21,800 m^3 (28,500 yd^3) 5,200 metric tons (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr (average), 234/yr (peak)
Duration (years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 22 m^3/yr (780 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.59 million l/yr (420,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

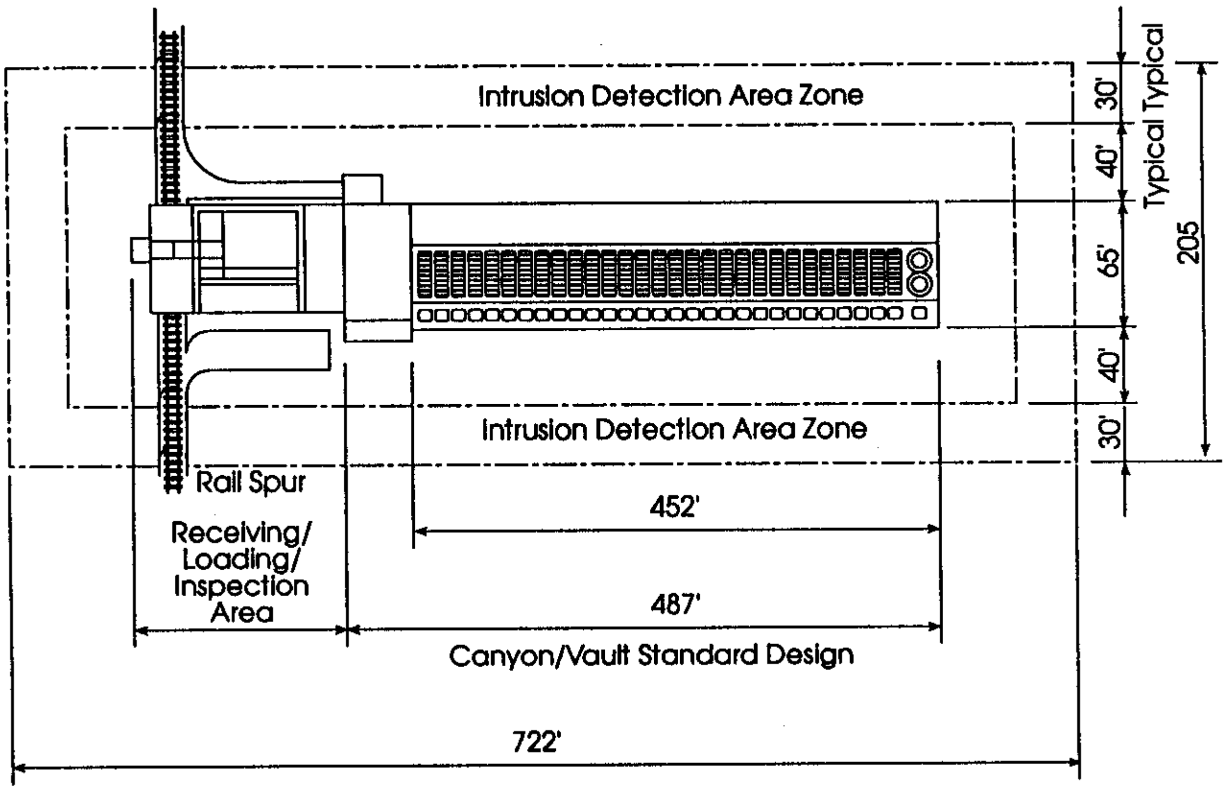
[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke, et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

area. Large, labyrinth air supply ducts and discharge chimneys would permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls would provide for shielding. The design would allow for expansion by adding additional units of arrays to the end of the vault or by construction of another vault. The vault facility would also include a receiving and loading bay that would allow handling of shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required and could be used for spent nuclear fuel characterization with additional equipment and modifications. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by the design. Table 2-8 summarizes modular dry vault storage parameters for foreign research reactor spent nuclear fuel storage.

In operation, the transportation cask would be lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements would be removed underwater, examined, and if the heat generation rate is below 40 Watts per element, the spent nuclear fuel would be placed within the transfer canister. The transfer canister would be subsequently drained, dried, and seal-welded. The handling machine then would place the spent nuclear fuel inside of the spent nuclear fuel storage canister, and would transport the loaded canister to the storage tubes. The handling machine would include radiation shielding. Heat dissipation would be accomplished by natural convection from the surfaces of the handling machine and canister. Decay heat would be dissipated by natural convection: air would enter through inlet ducts at the bottom of the vault module, pass around the outside of the steel storage tubes containing the spent nuclear

Note:
All dimensions shown are approximate and may only be used for a conceptual "footprint" estimate.



10-40 W/Element Scenario
25,000 FRR spent fuel element modular array (Nominal) Storage req. (29 modules with 44 tubes per module - 20 spent fuel elements per tube)

- Passive cooling
- Monitored storage canisters
- Retrievable for periodic inspection
- Self-contained facility for staging and storage of spent fuel elements

Figure 2-8 Layout of a Modular Dry Vault Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

fuel canisters, and exit through outlet ducts at the top of the module. Therefore, the vault would be a complete, integrated facility with all of the required capabilities for foreign research reactor spent nuclear fuel handling and storage.

The vault facility would store spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored within the canister in 5 levels with 4 elements per level, for a total of 20 spent nuclear fuel elements per canister (MTR-type design). The vault design would allow for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit nominally 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F). Thus, the number of vault units/arrays required for the storage of elements having a decay heat between 10 Watts and 40 Watts per element would be 27.

Most of the foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. For "cold" fuel (less than 10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. However, this would require a customized design, which could unnecessarily increase costs and implementation time. Figure 2-8 displays the layout of the modular dry vault storage facility (10 Watt to 40 Watt element basis).

Criticality concerns would be addressed primarily by the tube spacing in the vault. Borated concrete could also be used. For foreign research reactor spent nuclear fuel, criticality would not be expected to be a significant concern because a considerable fraction of the fissile uranium would have been consumed, and neutron-absorbing fission products would be present.

This vault design, without a pool, has been licensed by NRC for the Fort St. Vrain nuclear power plant site. It represents a complete, stand-alone facility that could be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, spent nuclear fuel transfer to a canister, and spent nuclear fuel storage could be accomplished within the facility. Additional facilities or modifications to the inspection area, including a pool, would be required for foreign research reactor spent nuclear fuel characterization.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

Spent Nuclear Fuel Storage Using Dry Casks:

Dry cask storage would include the use of concrete casks, both vertical and horizontal versions, metal casks, and multipurpose casks and would consist of the following components:

- A staging facility for cask receipt and unloading, and for loading foreign research reactor spent nuclear fuel into the dry storage casks. The staging facility would have a wet pool for unloading the casks and for short-term (1 to 3 years) storage of spent nuclear fuel with a heat load exceeding 40 Watts per element. This facility would include capabilities for drying the spent nuclear fuel/canister, inserting the spent nuclear fuel/canister with helium or nitrogen, and welding the storage canister closed.

- An inspection/characterization facility, for examining spent nuclear fuel integrity and canning leaking spent nuclear fuel as required. This may be incorporated into the staging facility (as an inspection cell) or be immediately adjacent to it. Although it is not expected that the physical condition of the foreign research reactor spent nuclear fuel elements would require extensive canning, the capability of canning the entire foreign research reactor spent nuclear fuel inventory would be provided by design.

- A dry storage cask (usually concrete). This would provide for the shielding and the structural stability of the spent nuclear fuel storage. The Multi-purpose Canister undergoing development could also be used (see Appendix F, Section F.1).

- A transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules or a crane for vertical modules.

- A separate spent nuclear fuel canister may or may not be used. If used, it would typically be approximately 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter, and would weigh approximately 33 metric tons (36 tons).

The dry cask approach would require the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, spent nuclear fuel would be loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask would be placed on a concrete slab located outdoors. The horizontal approach would use a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This would be placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram would insert the transfer canister inside the horizontal storage module, followed by sealing with a shield plug. Thus, dry cask storage would always rely on the use of another facility.

Dry storage casks would be designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. Concrete would provide radiation shielding for gamma rays and neutrons. Natural air circulation would dissipate the heat; air would enter through inlet vents near the bottom of the cask, pass around the spent nuclear fuel canister, and exit near the top. Screens and grills would keep birds and animals out of the cooling duct area.

Some of the potential management sites have facilities which could be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage. Utilization of these facilities would be considered.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel would depend upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load and are thus the focus of discussion. The standard design for a horizontal fuel canister would provide for 24 or 52 sleeves (i.e., pressurized water reactor or boiling water reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it would be conservatively assumed that each sleeve contains 5 foreign research reactor spent nuclear fuel elements (i.e., in layers) within a basket or can arrangement for maintaining spacing and retrievability. Also, as with the vault approach, the number of dry storage casks would depend upon the decay heat of the spent nuclear fuel and a cladding temperature limit [nominally, 175°C (347°F) for aluminum-cladding with an air inlet temperature of 49°C (120.2°F)]. The 24-sleeve design would allow for a maximum of 120 elements for foreign research reactor spent nuclear fuel with 40 Watts to 80 Watts per element of decay heat, while the 52-sleeve design would provide for a minimum of 260 elements per dry storage cask with 10 Watts to 40 Watts per element. Thus, based on the total number of elements for which the facilities are sized, the number of casks required would be:

- ninety-four casks, predicated upon a 3-year cooldown period (i.e., less than 40 Watts per element). Note that this value is conservative and corresponds to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Again, most foreign research reactor spent nuclear fuel is expected to have decay heats between 10 Watts and 40 Watts per element. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a final determination of limiting fuel storage temperature for aluminum-based and TRIGA-type spent nuclear fuel. However, the relatively high decay heat spent nuclear fuel represents such a small percentage of the currently identified foreign research reactor spent nuclear fuel that its impact would be small, such that after 3 years of wet storage, it would all be below a heat output of 40 Watts per element.

Figure 2-9 displays the general layout for the dry cask storage facility predicated upon a horizontal cask design. Table 2-9 summarizes dry cask storage parameters.

Dry storage cask technology would require a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading, and transportation cask maintenance. This facility would have the following operational areas:

- Transportation Cask Handling: this incorporates transportation cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.

- A Small Wet Storage Pool: for fuel transfer and short-term storage.

- Spent Nuclear Fuel Unit Handling: fuel removal, decontamination, fuel drying, fuel canning, inserting with helium, and thermal measurements.

- Spent Nuclear Fuel Unit Transfer: this constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.

- Radwaste Treatment: this includes collection, treatment, and preparation for disposal of contaminated effluents, and radioactive waste treatment and solidification.

- Heating, Ventilating, and Air Conditioning: this represents the component of the facility that helps ensure that contamination of workers and the environment is avoided.

The inspection/characterization facility would include a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of leaking spent nuclear fuel. All equipment and instrumentation within the cells would be remotely operated. The facility would be maintained under negative pressure with exhaust through high-efficiency particulate air filters to mitigate the environmental effects of any radionuclide releases. This facility is normally immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several management sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the Receiving

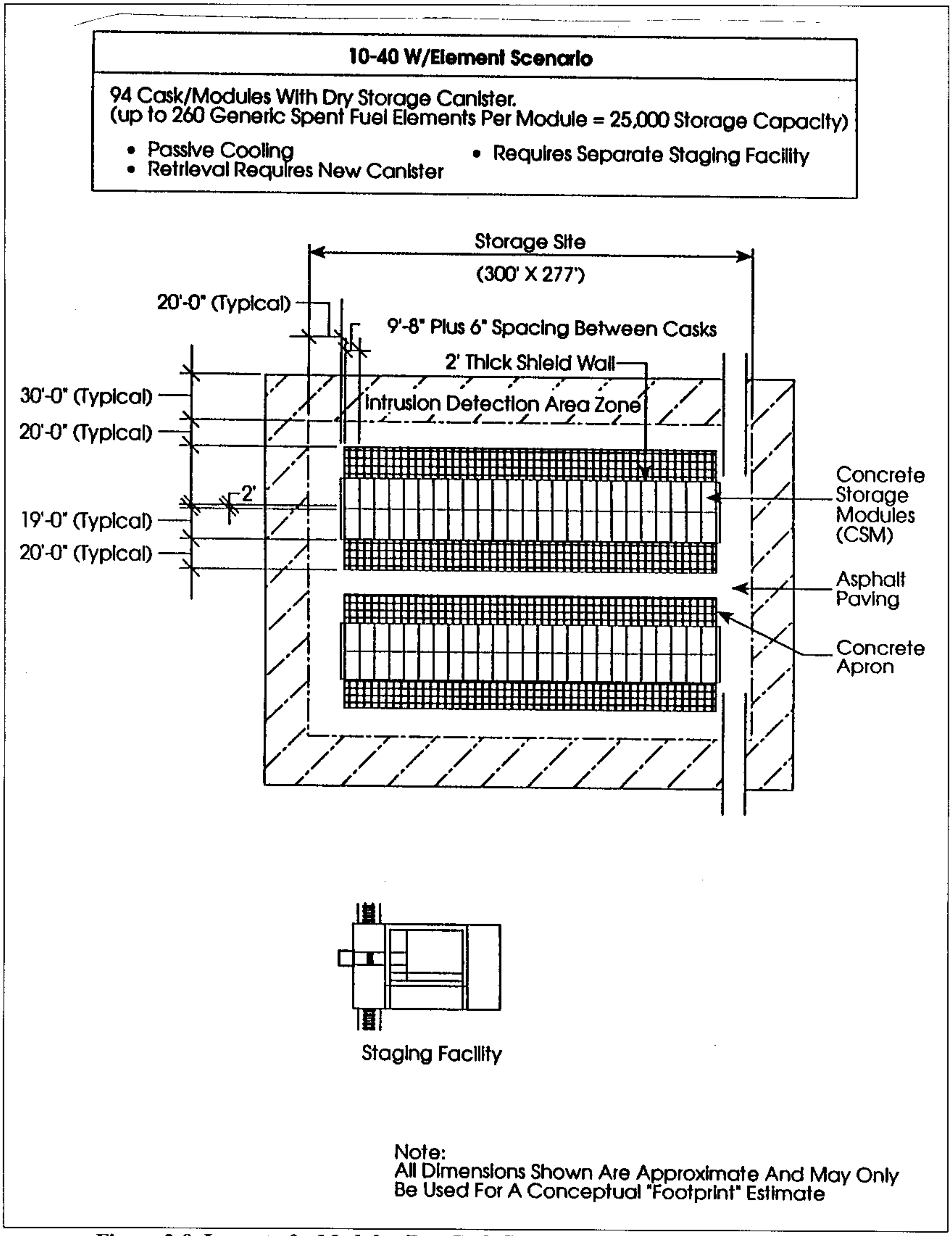


Figure 2-9 Layout of a Modular Dry Cask Storage Facility for Foreign Research Reactor Spent Nuclear Fuel (10 Watt to 40 Watt Element Basis)

Table 2-9 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: size (area) concrete steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility 50 per 24 cask array, 1 array per year
Duration (years)	5.5 for staging facility 4 years for construction, 1.5 years for design
Capital Cost	$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) during receipt 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams Solid Low-Level Waste Waste Water	 16 m^3/yr (565 ft^3/yr) during receipt 1 m^3/yr (35 ft^3/yr) thereafter 1.58 million l/yr (412,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt 8 thereafter
Annual Operating Cost	$17.3 million during handling, $0.3 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994)*

[b] *Cost estimates are in 1993 dollars (EG&G, 1993)*

Basin for Offsite Fuels (RBOF) at the Savannah River Site and the CPP-666 storage pool area at the Idaho National Engineering Laboratory. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a leaking element, and placed inside the storage canister. Spent nuclear fuel elements with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks).

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be $366 million. The annual operating cost for this facility is estimated to be $17.3 million during the period of handling and transfers of the spent nuclear fuel and $0.3 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

2.6.5.1.2 Description of Wet Storage Facilities

A wet storage facility consists of a spent nuclear fuel storage area and support areas (Dahlke et al., 1994). The spent nuclear fuel management area would provide for the receipt of cask transport vehicles, cask unloading and decontamination, and spent nuclear fuel handling, transfer, and storage. Support areas would provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilating, and air conditioning; water treatment; and waste management). The general layout of a wet storage facility is presented in Figure 2-10. The wet storage facility would be constructed as a safety class structure that meets all current nuclear regulations to withstand natural events such as seismic activity, tornadoes, and floods, as well as aircraft impact. Systems supporting the operation of the spent nuclear fuel management facility would also be required to meet these safety requirements. The facility would be equipped with a 118-metric ton (130-ton) overhead crane and a 9-metric ton (10-ton) spent nuclear fuel handling crane. Figure 2-11 displays a schematic of the facility, and Table 2-10 summarizes wet storage parameters for foreign research reactor spent nuclear fuel handling and storage.

Each cask transport vehicle would enter the facility through one of two bays where it would be monitored and washed to remove transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m by 24.4 m (10 ft by 80 ft) trucks or railcars, and casks weighing up to 114.3 metric tons (126 tons) each with a total individual cask and transport vehicle weight of 176 metric tons (195 tons). There are two unloading pools [6.1 m long and wide by 11.0 m deep (21 ft long and wide by 36 ft deep) and 6.4 m long by 5.8 m wide by 13.4 m deep (21 ft long by 19 ft wide by 44 ft deep)] and two decontamination rooms. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool [10.4 m long by 5.8 m wide by 9.4 m deep (34 ft long by 19 ft wide by 31 ft deep)] where it would be prepared for transfer to the storage pool buckets.

If cask measurements indicated that the spent nuclear fuel might be leaking, the spent nuclear fuel would be transferred to the isolation pool [3.7 m long by 3.0 m wide by 9.4 m deep (12 ft long by 10 ft wide by 31 ft deep)] for sipping. Sipping is a methodology for determining leaking spent nuclear fuel. This pool would be equipped so that wet sipping, dry sipping, or vacuum sipping of the suspect spent nuclear fuel element could be performed. An identified leaking spent nuclear fuel element would then be transferred to the cutting/canning pool where it would be canned before transfer to one of the storage pools. If it was not found to be leaking, it would be transferred directly to a storage pool.

All six pools in this facility (two unloading, two storage, one cutting/canning, and one leak check/isolation) would be hydraulically connected by a transfer channel/pool which would be 6.1 m long by 3.3 m wide by 9.4 m deep (20 ft long by 11 ft wide by 31 ft deep). Gates between this transfer channel and each pool would allow for hydraulic watertight isolation of the other pools. All pools and channels would be constructed of concrete with stainless steel floors and liners. Pool water leak detection and collection systems in accordance with NRC Regulatory Guide 1.13 (NRC, 1975) and American National Standards Institute, Standard N305-1975 (ANSI, 1975) would be provided for the pools.

Each of the two storage pools would be 16.5 m long by 10.4 m wide by 9.4 m deep (54 ft long by 34 ft wide by 31 ft deep), and each would contain 40 stainless steel storage racks. This would provide 1,000 storage holes with a 20 cm (8 in) spacing maintained between adjacent holes. The 20 cm (8 in) space provides neutron isolation between adjacent spent nuclear fuel elements, and would ensure criticality safety. Each rack would be 2.0 m square and 3.2 m high (6.7 ft square and 10.5 ft high) and consist of a

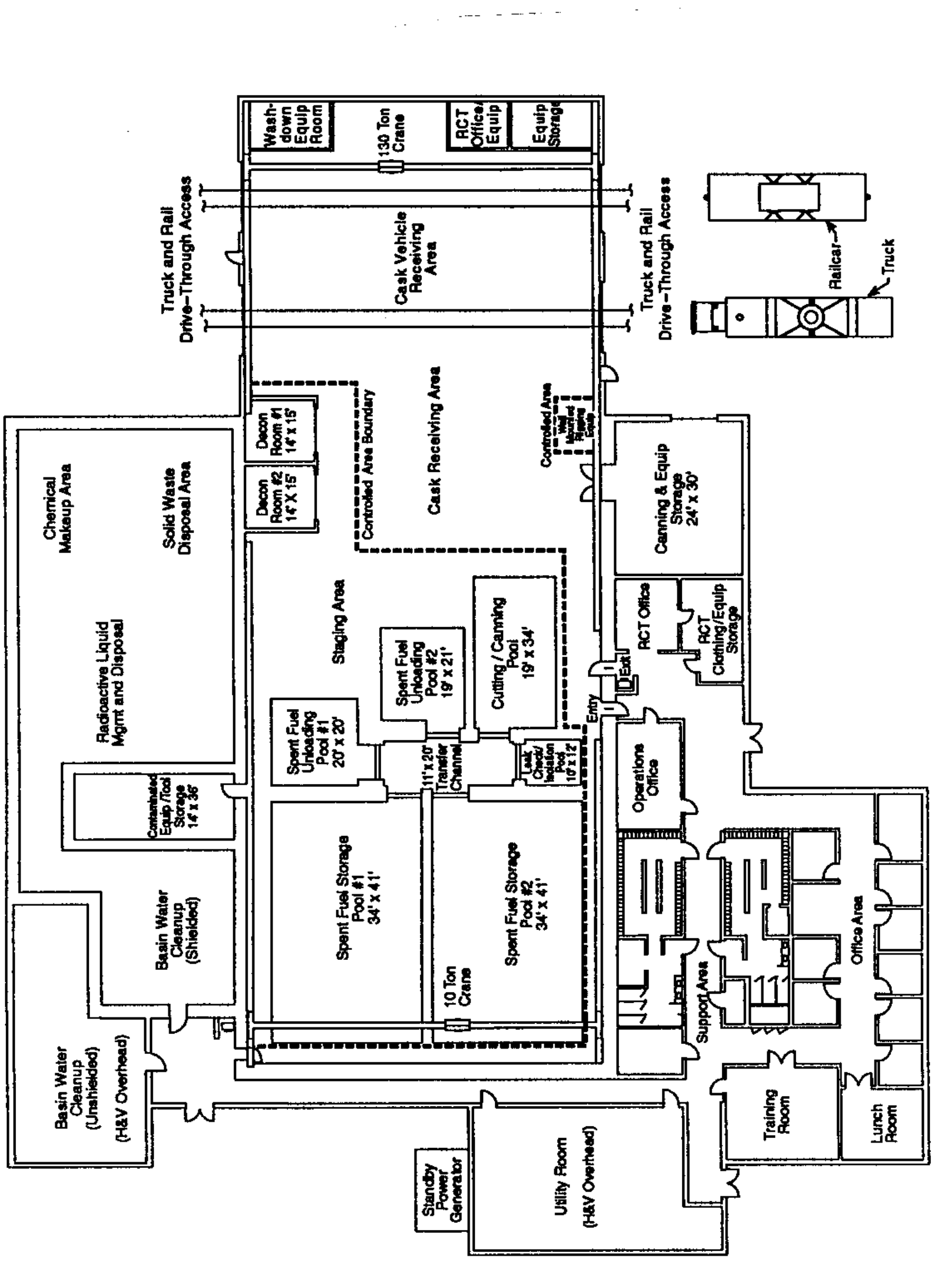


Figure 2-10 Generic Wet Storage Facility for Foreign Research Reactor Spent Nuclear Fuel

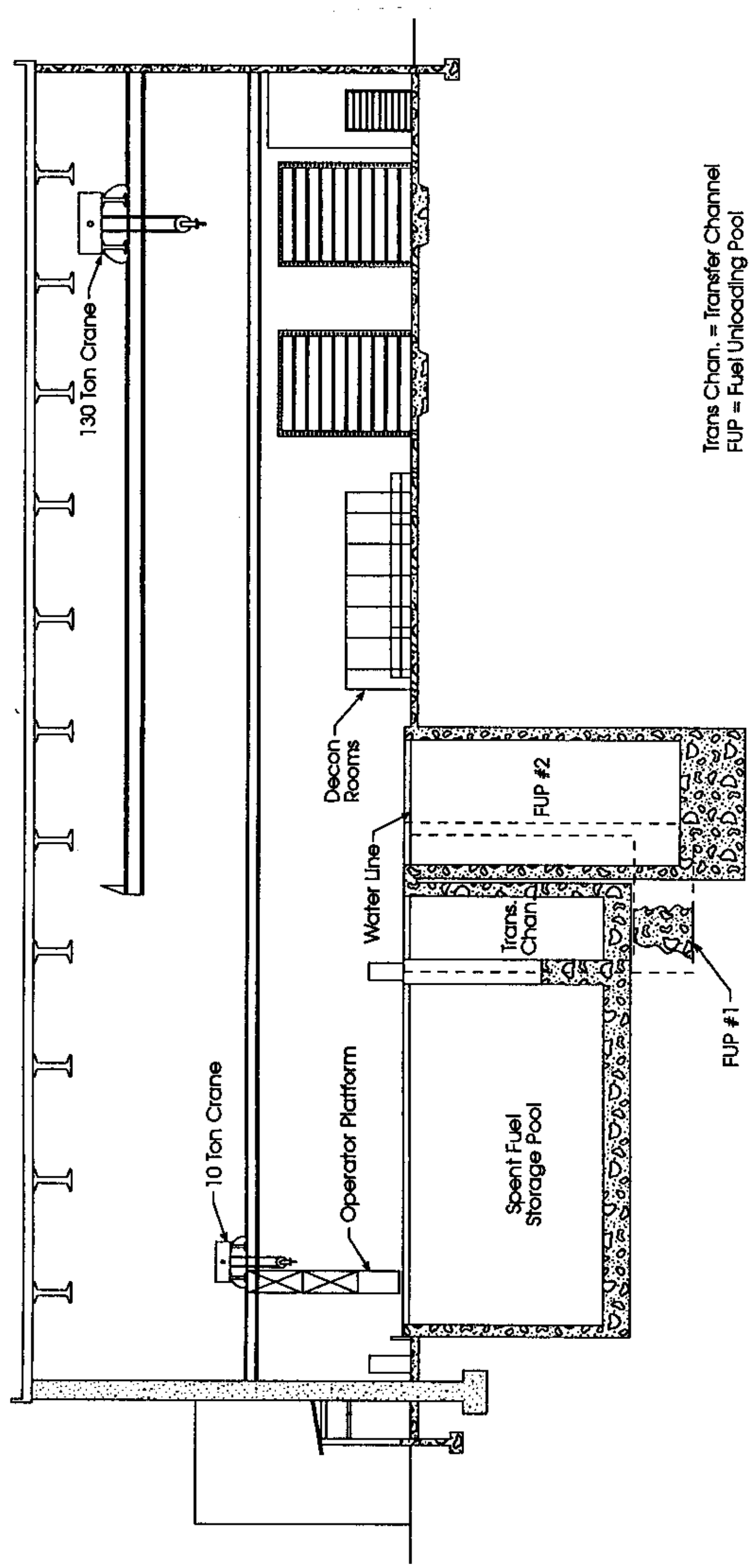


Figure 2-11 Schematic of a Wet Storage Facility for Foreign Research Reactor Spent Nuclear Fuel

Table 2-10 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: size (area) concrete steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr (average), 184 peak
Duration (years)	4 years for construction, 1.5 years for design
Cost	$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water (liters)	2.7 million l/yr (720,000 gal/yr) during receipt 1.5 million l/yr (409,000 gal/yr) thereafter
Waste Streams:	
High-Level Waste	none
TRU	none
Solid Low-Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Cost	$23.3 million during handling, $3.5 million during storage[b]

[a] *Cost estimates are in 1993 dollars (EG&G, 1993)*

[b] *The cost may include duplicate facilities and equipment present in both the staging and the rest of the wet storage facility.*

5 by 5 array of 25 spent nuclear fuel positions. A hinged lid would be above each of these spent nuclear fuel positions. Spent nuclear fuel elements would be stored in the racks so that at least 30 cm (12 in) of rack would protrude above the top of the fuel. Each position in the rack can hold up to three storage buckets, which would be stacked vertically on top of each other. The bucket, made up of 3.175 mm (0.125 in) thick stainless steel, would be fitted with ceramic spaces to prevent galvanic corrosion, and could store either two or four spent nuclear fuel elements, depending on the specific fuel design. This would provide a total capacity of approximately 12,000 elements for each storage pool.

The heating, ventilation, and air conditioning system for the wet storage facility would include a room of air supply equipment and a room for air exhaust equipment with separate filtering and monitoring. All exhaust air would be directed through pre-filters, high-efficiency particulate air filters, radiation monitors, filter fire protection components, and heat recovery coils before it would exhaust to the atmosphere.

The wet storage facility's water treatment system would consist of redundant pumps, piping, filters, deionizers and microorganism control systems. A heat removal system would be sized to maintain the bulk water temperature to acceptable levels. The system's filters and deionizers would include anion and cation exchangers that maintain water chemistry and remove radionuclides from the pool water.

The staff required to operate the wet storage facility would be a maximum of 30 when 24-hour-a-day fuel loading was being performed.

No high activity solid radioactive waste would be generated by the wet storage facility (equivalent to Class B or C low-level waste) over the life of the facility. Low-level solid radioactive waste that would be generated over the life of the facility would be about 640 m^3 (22,600 ft^3). Nonradioactive solid waste generated over the facility's life would be about 300 m^3 (10,600 ft^3). All ventilation air would pass through roughing and high-efficiency particulate air filters prior to exhaust. No nonradioactive, hazardous air emissions would be generated by this facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

2.6.5.2 Chemical Separation

Chemical separation involves separating the fissile material in the spent nuclear fuel from the other material (i.e., cladding material, fission products, etc.). Uranium and plutonium isotopes constitute the fissile materials; and with foreign research reactor spent nuclear fuel, relatively little plutonium and actinide elements are produced because the ^{238}U precursor is present in relatively small quantities. Waste materials would be mainly fission products (radioactive species such as cesium and strontium) in the form of liquid raffinates, low-level radioactive wastes, mixed radioactive/chemical wastes, waste acids, chelating and complexing agents, and organic solvents. The highly radioactive nature of fission products would require that the chemical separation activities be performed. Plutonium can be handled in facilities without radiation shielding, although these materials would still have to be handled under special procedures and precautions due to their radioactive, fissile nature. The other waste forms would require specialized handling, including volume reduction in some cases, to allow for safe storage and disposal.

Aqueous chemical methods are the only processing method applied on a large scale. All existing plants use an extraction process, which has been used for some 40 years. The spent nuclear fuel is initially dissolved in an acid and contacted with an organic solvent containing an extractant, such as tributylphosphate. The uranium and plutonium form a complex with the tributylphosphate and transfer to the organic phase. The cladding and waste materials remain in the aqueous phase, which is termed high-level waste. The uranium and plutonium are subsequently recovered by contact of the organic phase with weak acidic solutions. Vitrification of the high-level waste is the preferred waste management approach.

As discussed in Section 2.2.2.6 chemical separation is not a preferred technology for managing spent nuclear fuel in the United States.

Processing facilities exist at several DOE and foreign sites. The main domestic facilities are located at the Savannah River Site and Idaho National Engineering Laboratory. The main foreign facilities are in France and the United Kingdom.

Savannah River Site Facilities

At the Savannah River Site, two facilities are available to chemically separate the foreign research reactor spent nuclear fuel. These facilities are the F- and H-Canyons. The F- and H-Canyon facilities are nearly identical structures that use similar radiochemical processes for the separation and recovery of plutonium, neptunium, and uranium isotopes. The F-Canyon primarily recovered ^{239}Pu and ^{238}U from irradiated natural or depleted uranium, and the H-Canyon primarily recovered ^{238}Pu, ^{237}Np, and ^{235}U from irradiated reactor fuels and targets. The following paragraphs apply to both canyons unless noted.

The F- and H-Canyons are reinforced concrete structures, 255 m long by 37 m wide, and 20 m high (836.6 ft by 308 ft by 121.4 ft). They are named for the two areas ("canyons") in each structure that house the large equipment (tanks, process vessels, evaporators, etc.) used in the chemical separations processes performed in each facility. These areas are 170 m long by an average of 6 m narrow and 20 m deep (557.7 ft by 19.7 ft by 65.6 ft). The two canyons are parallel and open from floor to roof. A center section, which has four floors or levels, separates the canyons. The center section contains office space, the control room for all facility operations, chemical feed systems, and support equipment such as ventilation fans. Processing operations involving high radiation levels (dissolution, fission product separation, and high-level radioactive waste evaporation) occur in the "hot" canyon, which has thick concrete walls to shield people outside the facility and in the center section from radiation. The final steps of the chemical separations process, which generally involve lower radiation levels, occur in the "warm" canyon. Figure 2-12 shows the layout of F-Canyon.

Services typical for a large industrial facility are required to support the canyon operations. Such services include steam and cooling water for process vessels and a ventilation system.

A separate ventilation system serves portions of the facility, such as the hot and warm canyons, that contain the radioactive process equipment. This system ensures that the air pressure in such areas is below the pressure of the air outside the facility and the area occupied by workers. This design helps prevent the release of radioactive material outside the facility by ensuring that air always flows from the outside of the facility to the inside of the process areas. Air in the process areas is exhausted from the facility through a large filter that removes 99.5 percent of any airborne radioactive material from the air. A 61-meter-tall (200-ft) stack behind each canyon discharges this air to the atmosphere. This stack is the pathway for airborne emissions associated with the normal operation of the canyons.

Even though DOE has maintained the chemical separation facilities since their construction, they contain equipment and systems that have become degraded because of their age and changes in mission. In some cases the degraded condition of equipment can pose operational limitations. For example, at one time the H-Canyon contained equipment that provided the capability to dissolve not only aluminum-clad reactor fuel but also fuel clad in stainless steel. The electrolytic dissolver used for this purpose is no longer functional and has been abandoned in place.

Because of the ages of the facilities, they do not satisfy all current DOE requirements for the design and construction of nuclear facilities. For example, the canyons and associated B-Line facilities were built (during the Cold War when a primary concern was the potential for an attack) to resist a large external blast. The blast-resistant features of the canyons also make them resistant to such external natural phenomena as tornadoes and earthquakes. However, the canyons were not designed to withstand a severe earthquake (defined as producing a lateral ground acceleration that is 20 percent that of gravity or 0.2 g), as they would be if DOE were to build them today.

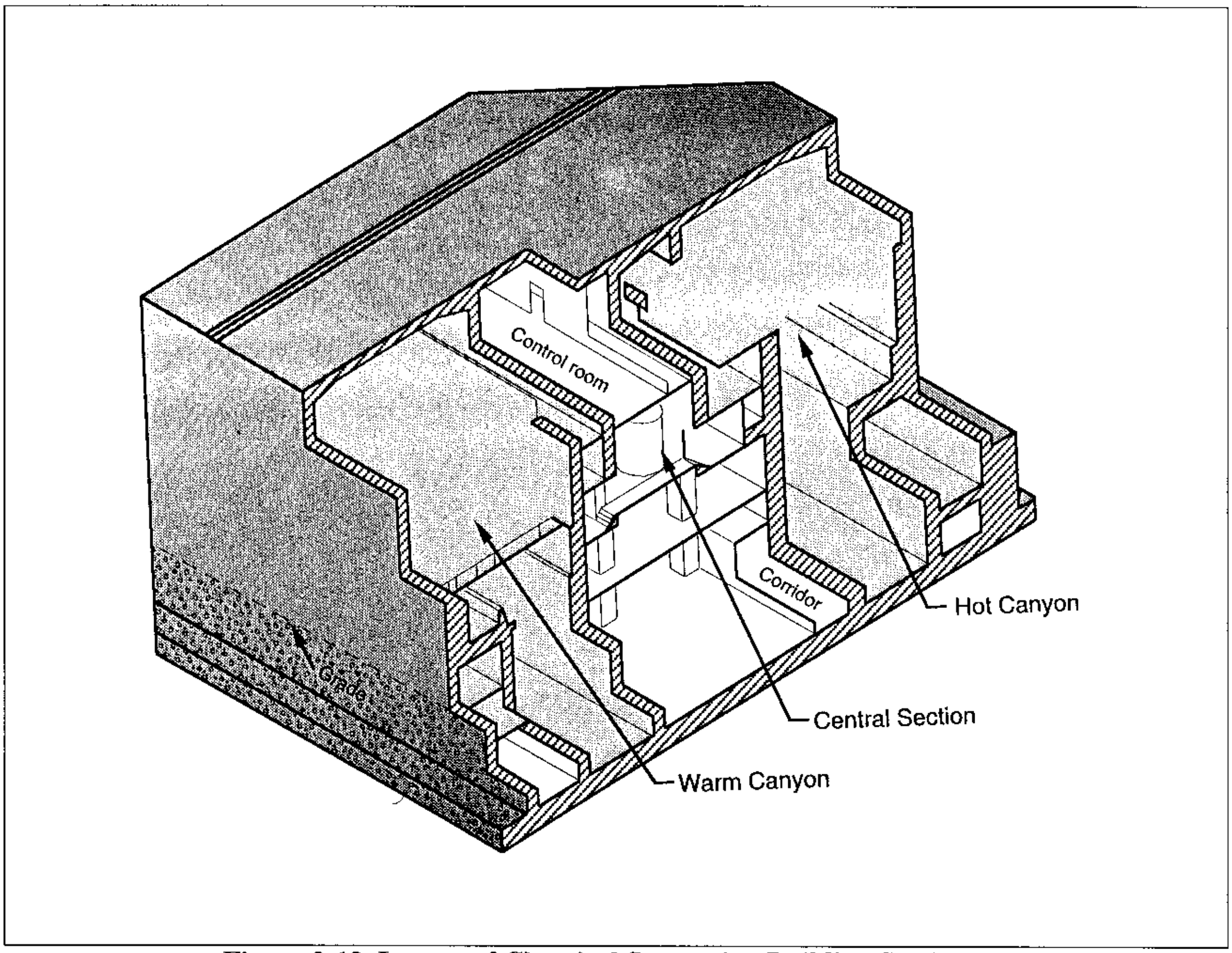


Figure 2-12 Layout of Chemical Separation Building Sections at Savannah River Site

The continued use of these facilities to chemically separate nuclear materials is an important factor for DOE consideration. Because the facilities do not meet current design and construction requirements, a facility-related vulnerability could produce environmental impacts (DOE, 1995a). As discussed above, the canyons would not maintain structural confinement of nuclear materials in a severe earthquake. The estimates of potential environmental consequences from accidents took this acknowledged vulnerability into consideration. If DOE were to design and construct a new facility, there would likely be no environmental consequences from a severe earthquake because a new facility would be designed to withstand such a force.

Similarly, in the Final Interim Management of Nuclear Materials EIS (DOE, 1995a), DOE considered other types of facility vulnerabilities in estimating the potential consequences from accidents. Some examples are (1) a fire that could spread in a facility until it breached containers of nuclear material due to a lack of detection or extinguisher systems, (2) systems that cool nuclear materials stored in tanks that could leak and transfer such material outside the facility before detection, or (3) piping configurations in the canyons that personnel could use inadvertently to transfer solutions of nuclear material to an outside facility tank where they could overflow or spill.

DOE has conducted many reviews to evaluate facility vulnerabilities and has assessed its facilities for compliance with current requirements. DOE has also analyzed the effect on workers and the public from normal and potential accident conditions which could result from operation of facilities with these vulnerabilities. The analysis work was accomplished as a part of ongoing safety review programs and is separate from the NEPA process. Such impact information is represented in the Final Interim Management of Nuclear Materials EIS and in this EIS. The analysis of impacts has, in some cases, prompted DOE to take corrective action based on potential impact alone. For example, DOE has disconnected some tanks of radioactive solutions in the canyons from the canyon cooling system and has isolated canyon tanks by removing interconnected piping to preclude leaks or an inadvertent transfer which could result in a release of radioactive material outside the canyon. In other cases, the potential impact was determined to be small and not sufficient to warrant actions beyond those which could be taken using existing facilities, equipment, and personnel. For example, one vulnerability common to many facilities is that the facility could sustain structural damage in the event of a severe earthquake. This type of earthquake has been estimated to occur once every several thousand years. It would be prohibitively expensive to modify facilities to ensure that no structural damage would occur from such an accident. Rather, DOE has provided mitigation for the consequences of such accidents using engineering safeguards, such as structurally reinforcing tanks, and administrative controls, such as limiting the amount of radioactive material that can be contained in a facility.

H-Canyon Process

The H-Canyon utilized a modified plutonium uranium extraction process (HM process). The HM process unit operations were dissolution, head end, first solvent extraction cycle, second uranium solvent extraction cycle, and second neptunium (or second actinide) solvent extraction cycle. Figure 2-13 shows the historic general H-Canyon process flow.

- Dissolution - Irradiated foreign research reactor spent nuclear fuel was brought into the hot canyon in water-filled casks and through an air lock by railcar. The spent nuclear fuel consists of HEU and LEU aluminum-based irradiated fuel. The spent nuclear fuel was removed from the casks and loaded into a dissolver tank. Heated nitric acid in the tank dissolved the foreign research reactor spent nuclear fuel, resulting in a solution containing enriched uranium, ^{237}Np, small quantities of plutonium, and fission products from the reactor irradiation process, and the cladding material. ^{237}Np should be insignificant in the chemical separation of foreign research reactor spent nuclear fuel.

- Head End - The head end process prepared the target solution for uranium, plutonium, and neptunium separation. First, gelatin was added to precipitate silica and other solid impurities. Then the solution was transferred to a centrifuge, where silica and other impurities were removed as waste, and the clarified product solution was adjusted with nitric acid and water. The wastestream generated from the head-end process was chemically neutralized and sent to high-level waste tanks.

- First Cycle - First cycle operation removed fission products and other chemical impurities, and separated the solution into two product streams for further processing. The chemical properties of acid/solvent/product solutions in contact with each other caused the fission products, the uranium, and the neptunium to separate from the solution containing plutonium.

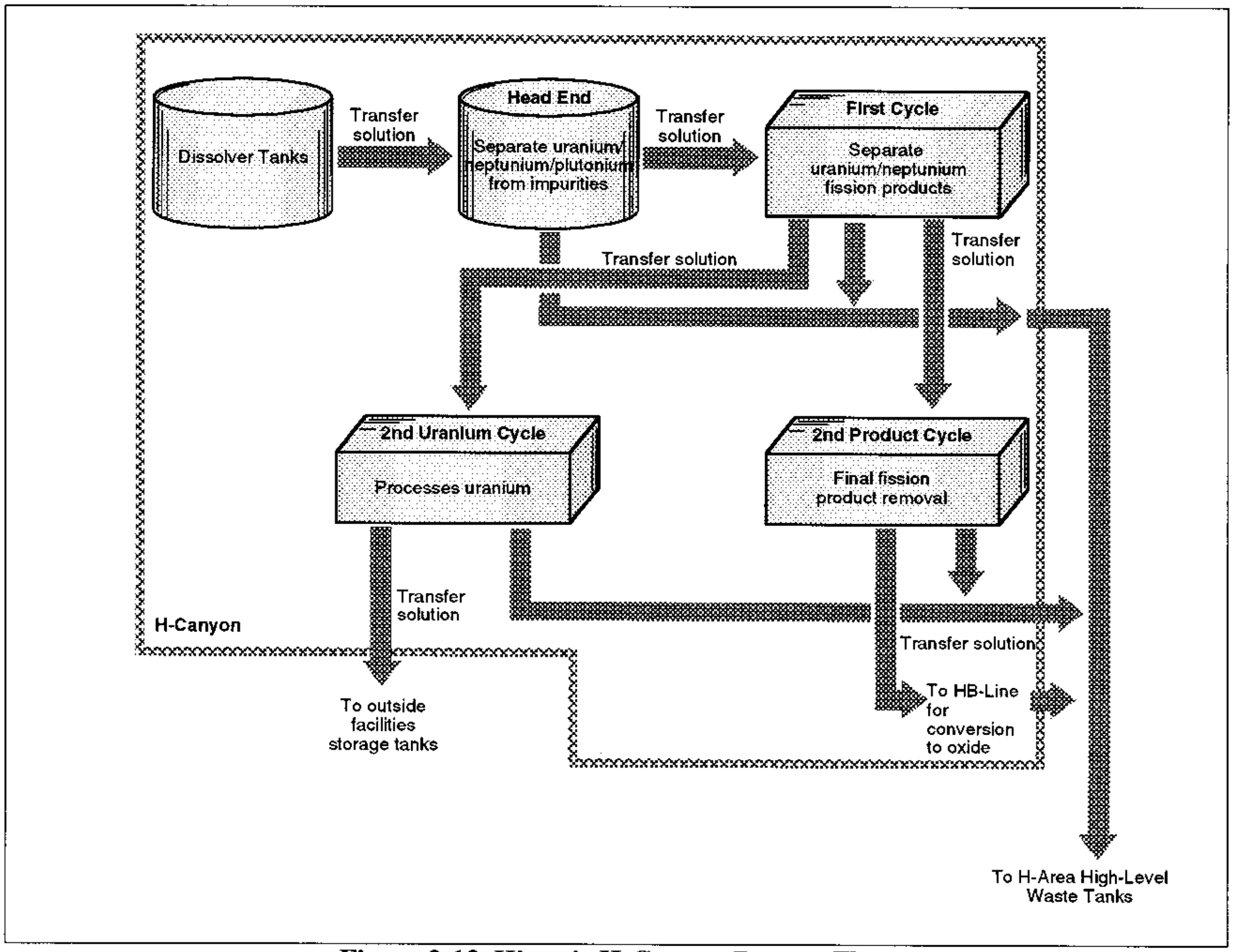


Figure 2-13 Historic H-Canyon Process Flow

- Second Uranium Cycle - The second uranium cycle purified the uranium solution from the first cycle and prepared the uranium for transfer. The purification occurred in a manner similar to that described for the first cycle. The ^{235}U product solution was transferred to storage tanks.

- Second Product Cycle - The second product cycle purified the neptunium solution from the first cycle by removing residual fission products, and prepare the neptunium for transfer. The process occurred in a manner similar to that for the first cycle. The impurities were removed and sent to the low-activity waste unit operation for processing. This cycle would probably be bypassed in the chemical separation of foreign research reactor spent nuclear fuel. Trace neptunium would be discarded as waste.

- High- and Low-Activity Waste - These unit operations reduced the volume of the aqueous streams containing fission products. The streams originate from the separation process unit operations, such as the first cycle. The fission product streams were then separated and sent to high-level waste tanks.

- Solvent Recovery - The primary purpose of this unit operation was to recover and recycle the solvent used in the first cycle. This operation reconditioned and removed impurities from the solvent. The purified solvent was returned to the first and second cycle, cycle reuse and the impurities were transferred to low-activity waste for processing.

F-Canyon Process

The Plutonium Uranium Extraction process at the F-Canyon includes unit operations such as dissolution, head end, first cycle, second uranium cycle, and second plutonium cycle. Unit operations that support the product recovery process were high-activity waste, low-activity waste, and solvent recovery. These were similar to those described for the HM process at H-Canyon with the exception of the Plutonium Uranium Extraction process. In the Plutonium Uranium Extraction process, the second plutonium cycle was equivalent to the second product cycle in the HM process.

Idaho National Engineering Laboratory Facilities

The Idaho Chemical Processing Plant (ICPP) facilities would use a Uranium Extraction process to chemically separate the foreign research reactor spent nuclear fuel for recovery of uranium, and isolation and solidification of the waste fission products resulting from the process. The principal facilities for foreign research reactor spent nuclear fuel chemical separation would be CPP-601, CPP-666, and CPP-602.

Foreign research reactor spent nuclear fuel would be received at the ICPP by truck or rail shipment. Both water-cooled and dry storage facilities would be used. Head-end equipment for initially dissolving or processing the spent nuclear fuel would be available. Aluminum-based clad fuel chemical separation could be conducted in CPP-601. TRIGA-type fuels would be processed in the Fluorinel Dissolution Process in CPP-666. The Fluorinel Dissolution Process cell could require some equipment modifications and additions to accommodate stainless steel-clad dissolution of the TRIGA fuel. However, the process knowledge and equipment is readily available. A new processing facility for uranium fuels is partially completed at the ICPP site. This facility, the Fuel Processing Restoration, is structurally complete but would require completion of services and installation of equipment for foreign research reactor spent nuclear fuel chemical separation.

The high-level liquid waste generated at ICPP during chemical separation of the spent nuclear fuel assemblies would be stored in several large stainless steel underground tanks until it could be processed. Liquid wastes would be converted to a solid calcine form, and then stored dry in bins housed in concrete vaults.

Aluminum and zircaloy fuel processing at ICPP consists of three principal stages. The first is the dissolution stage where fuels were dissolved forming a controlled solution. The second stage is the extraction process, which consists of first, second, and third extraction cycles. These cycles serve to separate and purify the uranium from fission products and material wastes prior to final operations. In the final operation, the solution is fed through a denitrator that conditions the feed material to a solid uranium product that can be packaged, transported, and recycled.

Foreign Reprocessing Facilities

Both France and the United Kingdom have modern fuel cycle facilities and offer reprocessing services to international customers. Either country could sign contracts with foreign research reactor owners/operators for receipt and reprocessing of their spent nuclear fuel, treatment of the waste, and fabrication of fresh fuel. Both France and the United Kingdom would require the country operating the reactor to take back the treated waste.

The French UP1 plant at Marcoule has reprocessed a variety of nuclear fuels, including gas/graphite power reactor fuel and magnesium-clad natural uranium metal fuel. The UP2 plant at La Hague is nearing completion of major renovations that will double its throughput and make it dedicated to oxide fuels. The UP3 plant, also at La Hague, is the newest French reprocessing plant. It started operations in 1990 and is also dedicated to oxide fuels. The French are vigorously engaged in reprocessing commercial power reactor fuel for foreign customers.

The British Prototype Fast Reactor Reprocessing Plant at Dounreay is a small plant associated with the Prototype Fast Reactor. However, it has established a precedent by receiving some research reactor spent nuclear fuel for reprocessing. The Magnox Fuel Reprocessing Plant at Sellafield reprocesses magnesium-clad uranium metal fuel from British gas-cooled reactors. The Thermal Oxide Reprocessing Plant (Thorp) is another large plant at Sellafield for Advanced Gas Reactor and light water reactor fuels. It started operating in January of 1994 and about two-thirds of its scheduled business through 2004 is for foreign customers.

2.6.5.3 Site Management Options

2.6.5.3.1 The Savannah River Site

Only two possible management sites, the Savannah River Site and the Idaho National Engineering Laboratory, would be capable of receiving and managing foreign research reactor spent nuclear fuel at the beginning of the proposed policy implementation period as described in Management Alternative 1.

If the Savannah River Site is the site for managing all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed there until ultimate disposition. If the Savannah River Site is not the site, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until another site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; modifications to existing facilities could take less. For the purposes of the analyses, the period for Phase 1 is assumed to be 10 years. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Savannah River Site under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and discussed in Section 2.6.4.1. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Eastern ports under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel (both aluminum-based and TRIGA) under the Centralization alternative.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at its existing wet storage facilities: RBOF and L-Reactor disassembly basin are considered for this purpose. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until the storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearranging and consolidating existing inventory. Descriptions of RBOF, the Savannah River Site reactor disassembly basins, and dry cask storage are provided in Appendix F, Section F.3.

The Savannah River Site reactor disassembly basins are not currently configured for storage of MTR type foreign research reactor spent nuclear fuel, however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements per reactor basin. DOE is considering the L-Reactor disassembly basin for this purpose in this EIS. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between RBOF and the L-Reactor disassembly basin there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year time period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities of RBOF and/or L-Reactor disassembly basin to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility. The storage capacity available and estimated maximum receipt rate of foreign research reactor spent nuclear fuel at the Savannah River Site are shown in Figure F-16 of Appendix F.

As a Phase 2 site, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The location is preferred among a number of sites considered as discussed in Section F.4.1. Foreign research reactor spent nuclear fuel managed during Phase 1 would be transferred to the new facility for management during Phase 2 (approximately 30 years), until ultimate disposition. The dry storage would encompass a number of design examples which were provided in Section 2.6.5.1.1 and Appendix F. Figure 2-14 depicts the facilities and locations considered at the Savannah River Site.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, are as follows:

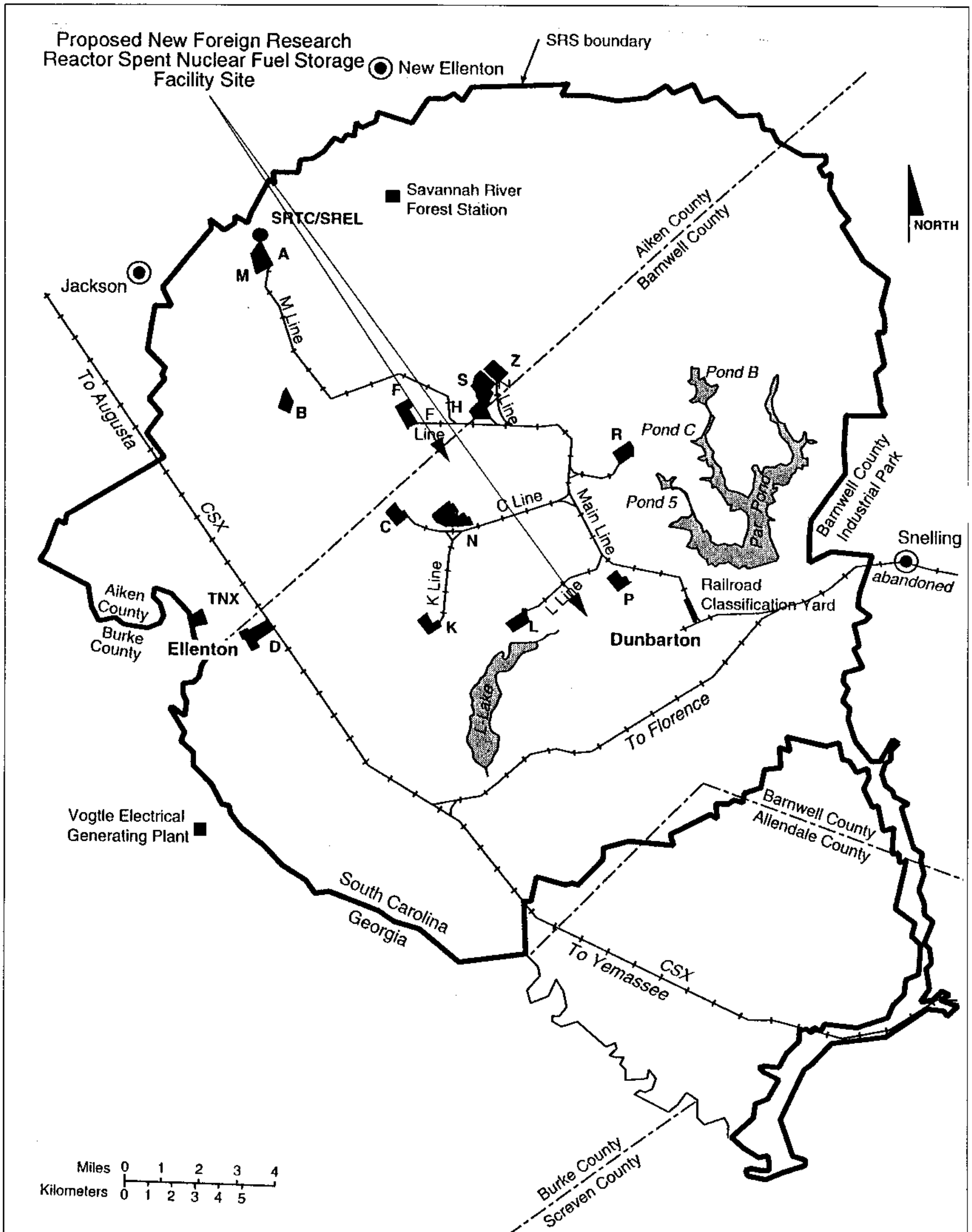


Figure 2-14 Location of Principal Facilities at the Savannah River Site

1A. The Savannah River Site would receive and manage foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of the analysis, the amount of fuel to be managed is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and managed at the new dry storage facility. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Savannah River Site.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which, in uranium content, represents approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 1A or 1B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A or 1B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 1A or 1B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States because the foreign research reactors would consider their own alternatives as to whether or not to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, is considered under analysis options 1A and 1B (above), which would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Savannah River Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider wet storage technology for Phase 2 management. DOE would implement this alternative by constructing a new wet storage facility at the H-Area or by using the Barnwell Nuclear Fuels Plant (BNFP), owned by Allied General Nuclear Services. DOE would have to acquire the facility which could be ready for use in approximately 5 years. Therefore, if the Savannah River Site was a selected site under either the Regionalization by Fuel Type or Centralization alternatives, Phase 2 at the Savannah River Site could start as early as 5 years from the start of the implementation period if BNFP were used under this implementation alternative. The new wet storage facility is described in Section 2.6.5.1.2. BNFP is described in Appendix F, Section F.1. For this implementation alternative, an analysis option 1C is considered, which is similar to 1B, as follows:

1C. The spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed wet storage facility or the BNFP where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would be received and managed at these facilities. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Processing Facility at the Savannah River Site. DOE and the Department of State estimate that the total volume of the vitrified high-level waste would be about 2.4 m^3 (85 ft^3) and it would fill about 16 European-size canisters. A European-sized canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site. Some modification to the waste handling facility at the Savannah River Site would be necessary to accommodate the smaller canisters.

Under Management Alternative 3 (Hybrid Alternative) discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as

Implementation Alternative 6 above. The amount of aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 72 m^3 (2,700 ft^3) as indicated in Table 2-4.

Table 2-11 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-11 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Savannah River Site

FRR EIS Management Alternative		FRR SNF Elements	Percentage of FRR SNF Total Elements	*Storage Option/Technology*					
				Dry Storage	*Wet Storage*			*Existing Facilities Plus Dry Cask*[c]	*Chemical Separation*
Management Alternative 1				*New*	*Existing*[a]	*BNFP*[b]	*New*		
All FRR SNF	Phase 1	17,500	77%	NA	A	NA	NA	A	NA
	Phase 2[d]	22,700	100%	A	NA	A	A	NA	NA
Eastern FRR SNF	Phase 1	12,600	56%	NA	A	NA	NA	A	NA
	Phase 2	16,400	72%	A	NA	A	A	NA	NA
Aluminum-based FRR SNF	Phase 1	13,600	60%	NA	A	NA	NA	A	NA
	Phase 2	17,800	78%	A	NA	A	A	NA	NA
Chemical Separation/Storage	Phase 2	17,800	78%	NA	A	NA	NA	A	A
Management Alternative 3									
Aluminum-Based FRR SNF Chemical Separation/Storage		12,300	54%	NA	A	NA	NA	A	A

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *RBOF and L-Reactor basin*
[b] *BNFP could be available for use 5 years after the start of implementation.*
[c] *Dry cask storage would use an existing facility for loading operations.*
[d] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.2 Idaho National Engineering Laboratory

Only two possible management sites, the Savannah River Site and the Idaho National Engineering Laboratory, would be capable of receiving and managing foreign research reactor spent nuclear fuel at the beginning of the proposed policy implementation period.

If the Idaho National Engineering Laboratory is the site for managing all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed there until ultimate disposition. If the Idaho National Engineering Laboratory is not the site, foreign research reactor spent nuclear fuel could be received and managed at the Idaho National Engineering Laboratory until another site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; this period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, the Idaho National Engineering Laboratory could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the TRIGA-type foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Western ports under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

As a potential Phase 1 site, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and wet storage facilities. The existing facilities identified for this purpose would be the Fluorinel Dissolution and Fuel Storage (FAST) facility in CPP-666, the Irradiated Fuel Storage Facility (IFSF) in CPP-603, and the CPP-749 storage area. Descriptions of these facilities are provided in Appendix F, Section F.3.

The FAST facility is a modern underwater storage facility which has been used in the past for receipt and storage of foreign research reactor spent nuclear fuel. It has the capability to receive and unload spent nuclear fuel casks at a rate of approximately five per week. Storage capacity for up to 8,400 foreign research reactor spent nuclear fuel elements could be provided in a 10-year period by using the spent nuclear fuel storage racks that would be installed. The capability of the FAST facility to receive foreign research reactor spent nuclear fuel in the near term is limited due to the number of activities scheduled through FY 1998. Considering these activities, DOE estimates that 3,600 elements could be received by the end of 1999 at the FAST facility.

The IFSF is a shielded dry storage vault originally constructed for Fort St. Vrain reactor fuel. The storage capacity available is for approximately 9,000 foreign research reactor spent nuclear fuel elements. However, as with the FAST facility, many activities are already scheduled for the facility. Considering these activities, foreign research reactor spent nuclear fuel could not be received until sometime in FY 1997 and could continue at the rate of 50 shipments per year (approximately 1,500 elements) thereafter.

The CPP-749 underground spent nuclear fuel storage area is a dry storage facility with a remote unloading area and vault storage. With some refurbishment it could provide space for 3,600 elements starting in FY 1998. The spent nuclear fuel would go through the IFSF to be placed in baskets and transferred to a compatible storage cask. The refurbishments of existing facilities are part of the ongoing programs at the site, to be performed independent of the proposed action in this EIS.

Between these facilities there is sufficient storage space and handling capacity to accommodate the receipt and management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory during the Phase 1 period. The storage capacity available and estimated maximum receipt rate of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory are shown in Figure F-18 of Appendix F.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility. Descriptions of the Idaho National Engineering Laboratory existing facilities are provided in Appendix F. The location of these facilities at the Idaho National Engineering Laboratory are shown in Figure 2-15.

As a Phase 2 site, the Idaho National Engineering Laboratory would continue to receive and manage foreign research reactor spent nuclear fuel at existing facilities until a new dry storage facility were to become operational at the site. Foreign research reactor spent nuclear fuel managed at existing facilities

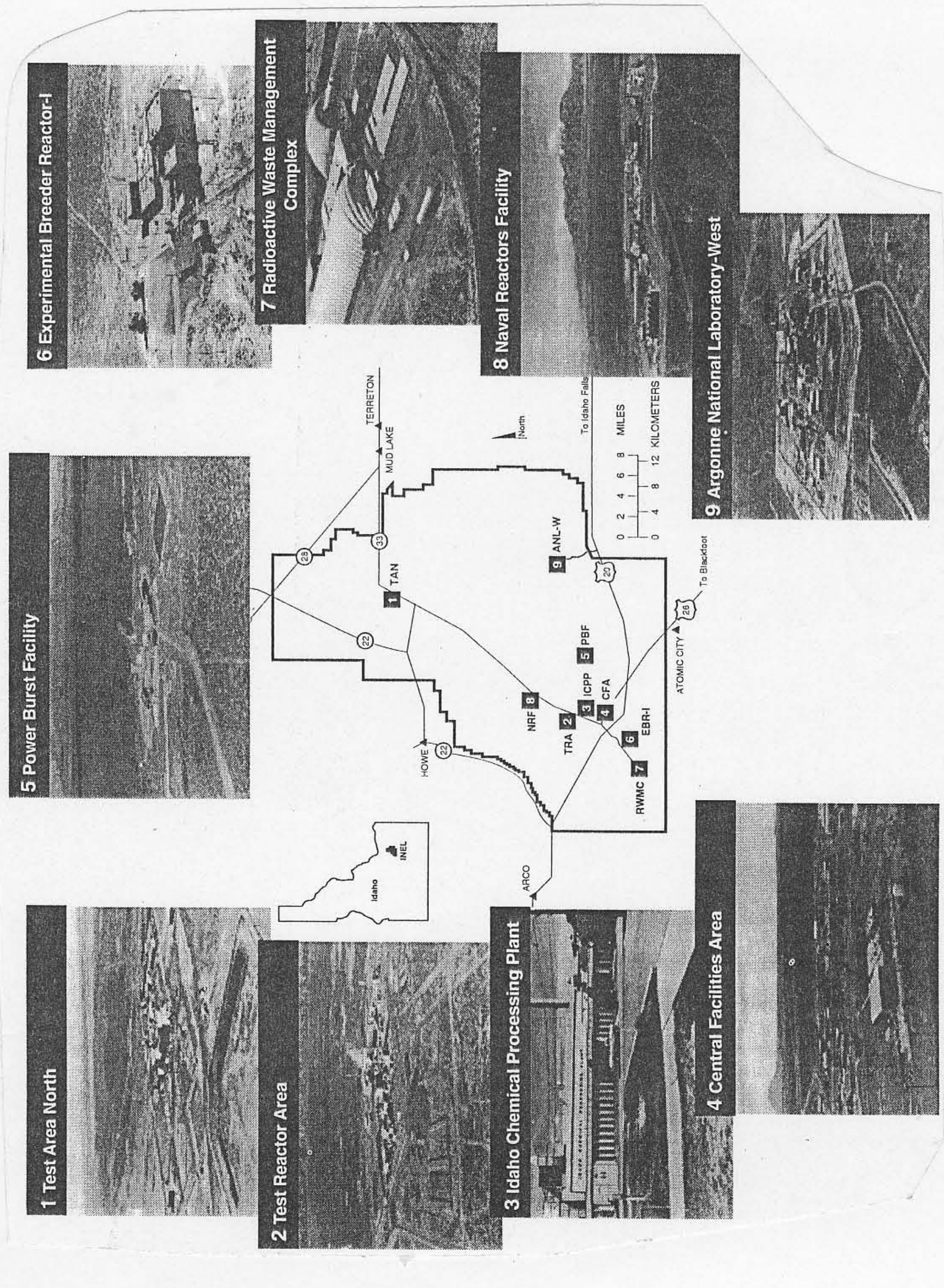


Figure 2-15 Location of Principal Facilities at the Idaho National Engineering Laboratory

would then be transferred to the new facility where it would remain until ultimate disposition. The new facility would also receive foreign research reactor spent nuclear fuel shipments directly from ports after the 10-year policy period. Dry storage encompasses both the dry vault design and the dry cask design as described in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, are as follows:

2A. The Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel during Phase 1 and store it at the FAST, the IFSF, and/or the CPP-749 facilities. For the purpose of the analysis, the amount of fuel to be managed is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

2B. Foreign research reactor spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed dry storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving at the United States after Phase 1 concludes would be received and managed at the new dry storage facility until ultimate disposition. For the purpose of the analysis, the amount of spent nuclear fuel that would be stored in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1 discussed in Section 2.2.2 introduce additional analysis options that could be considered for the Idaho National Engineering Laboratory as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Idaho National Engineering Laboratory would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 2A above. The dry storage facility considered in analysis option 2B would be sized to accommodate this amount of fuel. The fuel would either be shipped offsite after Phase 1 or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Idaho National Engineering Laboratory.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive only HEU from the reactors eligible under the proposed action. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A and 2B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents, in uranium content, approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 2A or 2B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and therefore the amount of spent nuclear fuel available for management would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A or 2B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 2A or 2B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States because the foreign research reactors would consider their own alternatives about whether to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, is considered under analysis options 2A or 2B which would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Idaho National Engineering Laboratory.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Idaho National Engineering Laboratory for Phase 2 until ultimate disposition. The new wet storage facility is described in Section 2.6.5.1.2. For this implementation alternative, an analysis option 2C, which is similar to analysis option 2B, is considered as follows:

2C. The spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed wet storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would be received and managed at the new wet storage facility until ultimate disposition. For the purpose of the analysis, the amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in the discussion in Section 2.3.6, chemical separation of both aluminum-based and TRIGA foreign research reactor spent nuclear fuel is evaluated for the Idaho National Engineering Laboratory.

Under Management Alternative 3 (Hybrid Alternative), discussed in Section 2.4, the Idaho National Engineering Laboratory would receive the foreign research reactor TRIGA spent nuclear fuel. This spent nuclear fuel would be managed at the Idaho National Engineering Laboratory in existing facilities until final disposition. The amount of TRIGA spent nuclear fuel to be stored would be 4,900 elements, 1.0 MTHM, and about 4 m^3 (200 ft^3) as indicated in Table 2-4.

Table 2-12 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-12 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Idaho National Engineering Laboratory

FRR EIS Management Alternative		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*					
				Dry Storage		*Wet Storage*		*Existing Facilities plus Casks*[c]	*Chemical Separation*
Management Alternative 1				*New*	*Existing*[a]	*New*	*Existing*[b]		
All FRR SNF	Phase 1	17,500	77%	NA	A	NA	A	A	NA
	Phase 2[d]	22,700	100%	A	NA	A	NA	NA	NA
Western FRR SNF	Phase 1	4,800	21%	NA	A	NA	A	A	NA
	Phase 2	6,300	28%	A	NA	A	NA	NA	NA
TRIGA FRR SNF	Phase 1	3,800	17%	NA	NA	NA	A	A	NA
	Phase 2	4,900	22%	NA	A	A	NA	NA	NA
Chemical Separation/Storage		22,700	100%	NA	A	NA	A	A	A
Management Alternative 3									
TRIGA FRR SNF/ Storage		4,900	22%	NA	A	NA	A	A	NA

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *IFSF, CPP-749*
[b] *FAST*
[c] *Existing facilities augmented by dry cask storage.*
[d] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.3 The Hanford Site

Options for receiving and managing foreign research reactor spent nuclear fuel at the Hanford Site are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives, and the lack of suitable facilities at the Hanford Site to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Hanford Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Hanford Site to have new facilities constructed and operational to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Hanford Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National

Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Hanford Site until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Hanford Site under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, in Phase 2, the Hanford Site could receive the TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

Under the basic implementation of Management Alternative 1, and as a Phase 2 site, the Hanford Site would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility constructed at the 200 Area Plateau or the Fuel Material Examination Facility (FMEF), which is a partially completed, large, hot cell facility. The new dry storage facility is described in Section 2.6.5.1.1. Description of the FMEF is provided in Appendix F, Section F.3. Figure 2-16 displays the potential construction locations for foreign research reactor spent nuclear fuel storage at the Hanford Site. FMEF is located near the Fast Flux Test Facility.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Hanford Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

3A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new dry storage facility constructed either at the 200 Area Plateau or at FMEF. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Hanford Site were to receive TRIGA from the Idaho National Engineering Laboratory or only western spent nuclear fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

The implementation alternatives of Management Alternative 1, which are discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Hanford Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Hanford Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and manage it in facilities sized for this amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 3A.

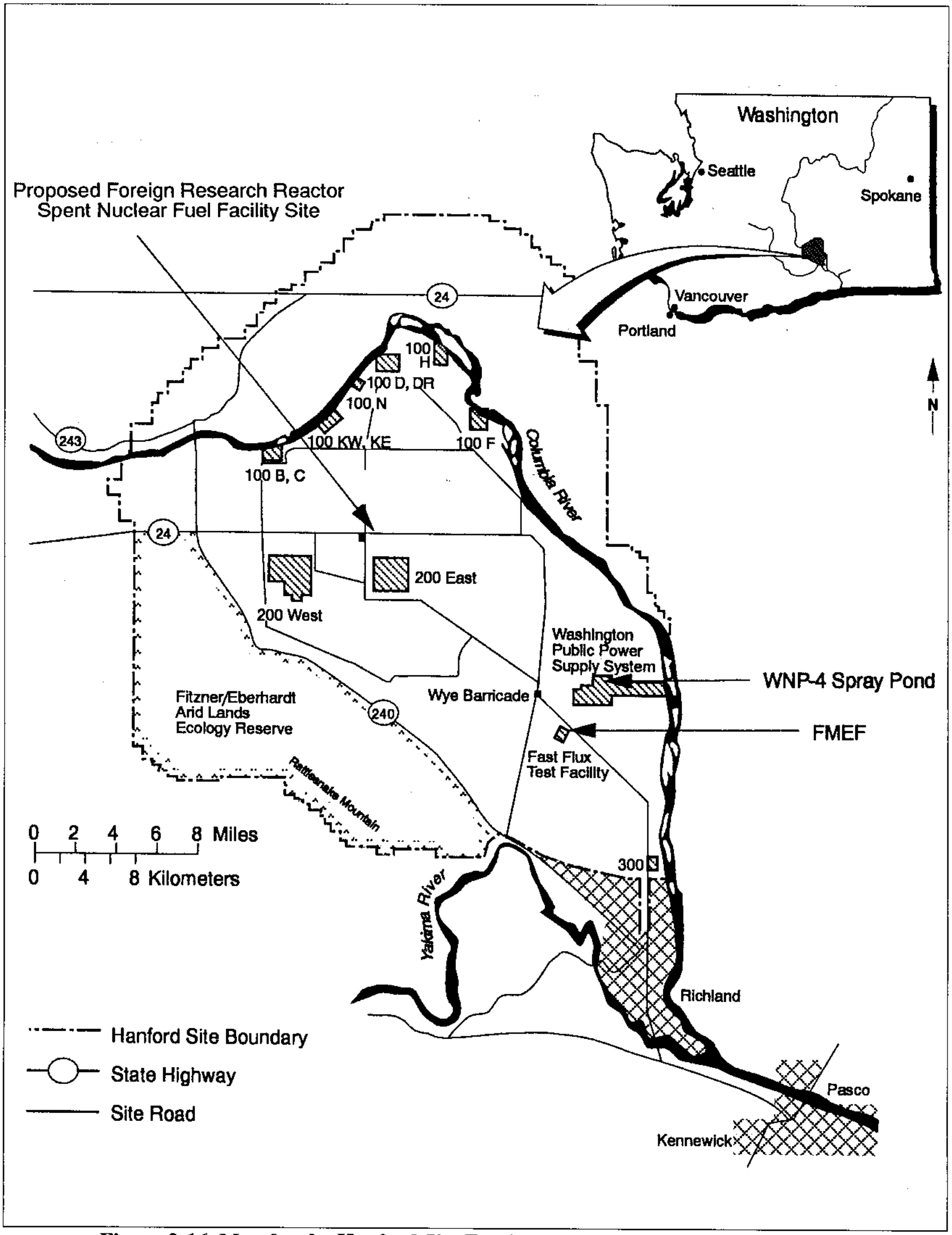


Figure 2-16 Map for the Hanford Site Foreign Research Reactor Spent Nuclear Fuel Storage (in the 200 Areas)

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Hanford Site would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Hanford Site would be bounded by analysis option 3A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Hanford Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 3A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In this case, the Hanford Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Hanford Site would be bounded by analysis option 3A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 3A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, which is considered under analysis option 3A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Hanford Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Hanford Site for Phase 2 until ultimate disposition. For this implementation alternative, an analysis option 3B, which is similar to 3A, is considered as follows:

3B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 is shipped to the Hanford Site where it would be managed at a new wet storage facility constructed at either the 200 Area Plateau or the WNP-4 Spray Pond facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Hanford Site were to receive only TRIGA fuel from the Idaho National Engineering Laboratory, or only western fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Hanford Site would not be considered as a site for chemical separation. The Hanford Site is also not considered under the Hybrid Alternative discussed in Section 2.4.

Table 2-13 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-13 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at Hanford Site

Management Alternative 1		*FRR SNF Element*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*			
				Dry Storage		*Wet Storage*	
				New at FMEF	*New*	*New at WNP-4*	*New*
All FRR SNF	Phase 2	22,700	100%	A	A	A	A
Western FRR SNF	Phase 2	6,300	28%	A	A	A	A
TRIGA FRR SNF	Phase 2	4,900	22%	A	A	A	A

A = Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel

2.6.5.3.4 The Oak Ridge Reservation

The options for receiving and managing foreign research reactor spent nuclear fuel at the Oak Ridge Reservation are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives and the lack of suitable facilities at the Oak Ridge Reservation to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Oak Ridge Reservation is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Oak Ridge Reservation to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Oak Ridge Reservation would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Oak Ridge Reservation until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Oak Ridge Reservation under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, in Phase 2, the Oak Ridge Reservation could receive aluminum-based foreign research reactor spent nuclear fuel managed at the Savannah River

Site during Phase 1, Eastern foreign research reactor spent nuclear fuel under the Regionalization by Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

Under the basic implementation of Management Alternative 1, and as a Phase 2 site, the Oak Ridge Reservation would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility to be constructed at the West Bear Creek Valley Site. The location is preferred among the four locations considered in a siting study performed for spent nuclear fuel management (MMES, 1994). The four locations considered are shown in Figure 2-17. Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation is based on the above considerations. The analysis options selected do not represent all possible combinations but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

4A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new dry storage facility until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, which are discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Oak Ridge Reservation as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Oak Ridge Reservation would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for this amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 4A.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Oak Ridge Reservation would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount of spent nuclear fuel would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Oak Ridge Reservation would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The analysis assumes that the receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 4A by a small percentage.

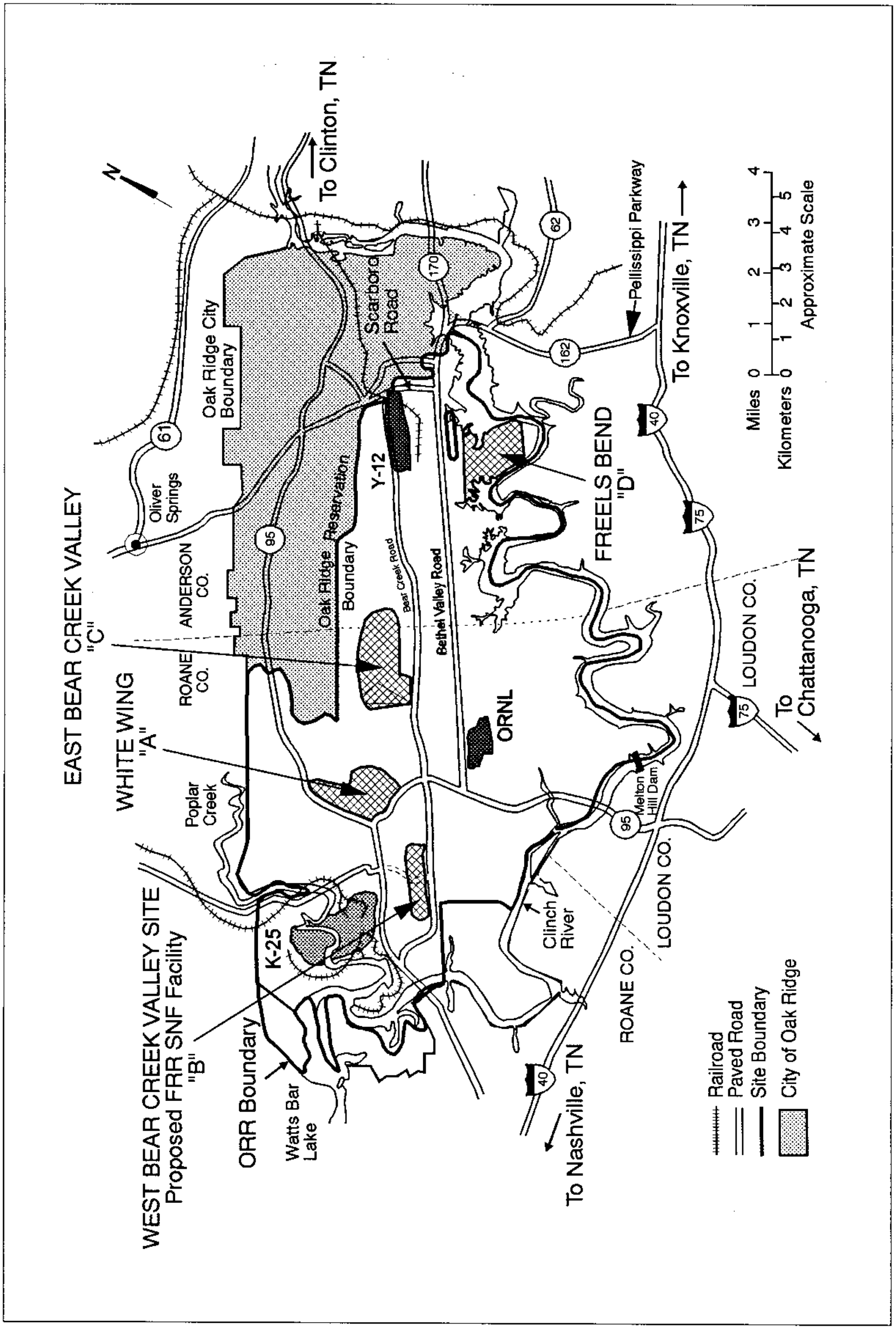


Figure 2-17 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In this case, the Oak Ridge Reservation would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 4A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, which is considered under analysis option 4A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Oak Ridge Reservation.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Oak Ridge Reservation for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 4B, which is similar to 4A, is considered as follows:

4B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Oak Ridge Reservation would not be considered as a site for chemical separation. The Oak Ridge Reservation is also not considered under the Hybrid Alternative discussed in Section 2.4.

Table 2-14 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-14 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at Oak Ridge Reservation

Management Alternative 1		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*	
				Dry Storage (New)	*Wet Storage (New)*
All FRR SNF	Phase 2[a]	22,700	100%	A	A
Eastern FRR SNF	Phase 2	16,400	72%	A	A
Aluminum-based FRR SNF	Phase 2	17,800	78%	A	A

A = Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.6.5.3.5 The Nevada Test Site

The options for receiving and managing foreign research reactor spent nuclear fuel at the Nevada Test Site are primarily dictated by the Programmatic SNF&INEL Final EIS (DOE, 1995c) alternatives, and the lack of suitable facilities at the Nevada Test Site to receive foreign research reactor spent nuclear fuel at the beginning of the proposed policy period.

If the Nevada Test Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period (Phase 1) required for the Nevada Test Site to have new facilities constructed and operational to accommodate the spent nuclear fuel. As discussed in previous sections, Phase 1 is estimated to be about 10 years. At the end of Phase 1 (i.e., start of Phase 2) the Nevada Test Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Nevada Test Site until ultimate disposition.

Although the Nevada Test Site has no existing facilities to receive foreign research reactor spent nuclear fuel at the beginning of the management period, it has facilities that could be modified to receive foreign research reactor spent nuclear fuel within 5 years. These facilities are large hot cells located in the Nevada Research and Development Area on Jackass Flats. Presently these facilities (e.g., the Engine Maintenance and Disassembly [E-MAD] facility) have little usage, but some are in acceptable condition. To use the E-MAD facility, a small pool would have to be constructed to be used for transferring the spent nuclear fuel from the transportation casks to containers designed for dry storage. A description of the E-MAD facility is included in Appendix F (Section F.1). The E-MAD facility could be ready within 5 years of the start of the proposed policy period.

The amount of spent nuclear fuel that would be received and managed at the Nevada Test Site under the basic implementation of Management Alternative 1, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS (DOE, 1995c). Accordingly, during Phase 2, the Nevada Test Site could receive TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, only Western foreign research reactor spent nuclear fuel under the Regionalization By Geography alternative, or all foreign research reactor spent nuclear fuel under the Centralization alternative.

As a Phase 2 site, the Nevada Test Site would receive and manage foreign research reactor spent nuclear fuel at a newly constructed dry storage facility or a modified E-MAD facility. Description of the new dry storage facility is provided in Section 2.6.5.1.1. Figure 2-18 displays the potential construction location and the area where the E-MAD facility is located at the Nevada Test Site.

The analysis of potential environmental impacts from management of foreign research reactor spent nuclear fuel at the Nevada Test Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis option under the basic implementation of Management Alternative 1, discussed in Section 2.2.1, is as follows:

5A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new dry storage facility or a modified E-MAD facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new or E-MAD facility until ultimate disposition. For the purposes of the analysis, the total amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel (22,700 elements).

The implementation alternatives of Management Alternative 1, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Nevada Test Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Nevada Test Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for the reduced amount. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 5A.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Nevada Test Site would receive from the Idaho National Engineering Laboratory and/or the Savannah River Site only HEU. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel would be bounded by analysis option 5A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Nevada Test Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 5A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years; and, therefore, the amount of spent nuclear fuel available for management would also be decreased. In such a case, the Nevada Test Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the

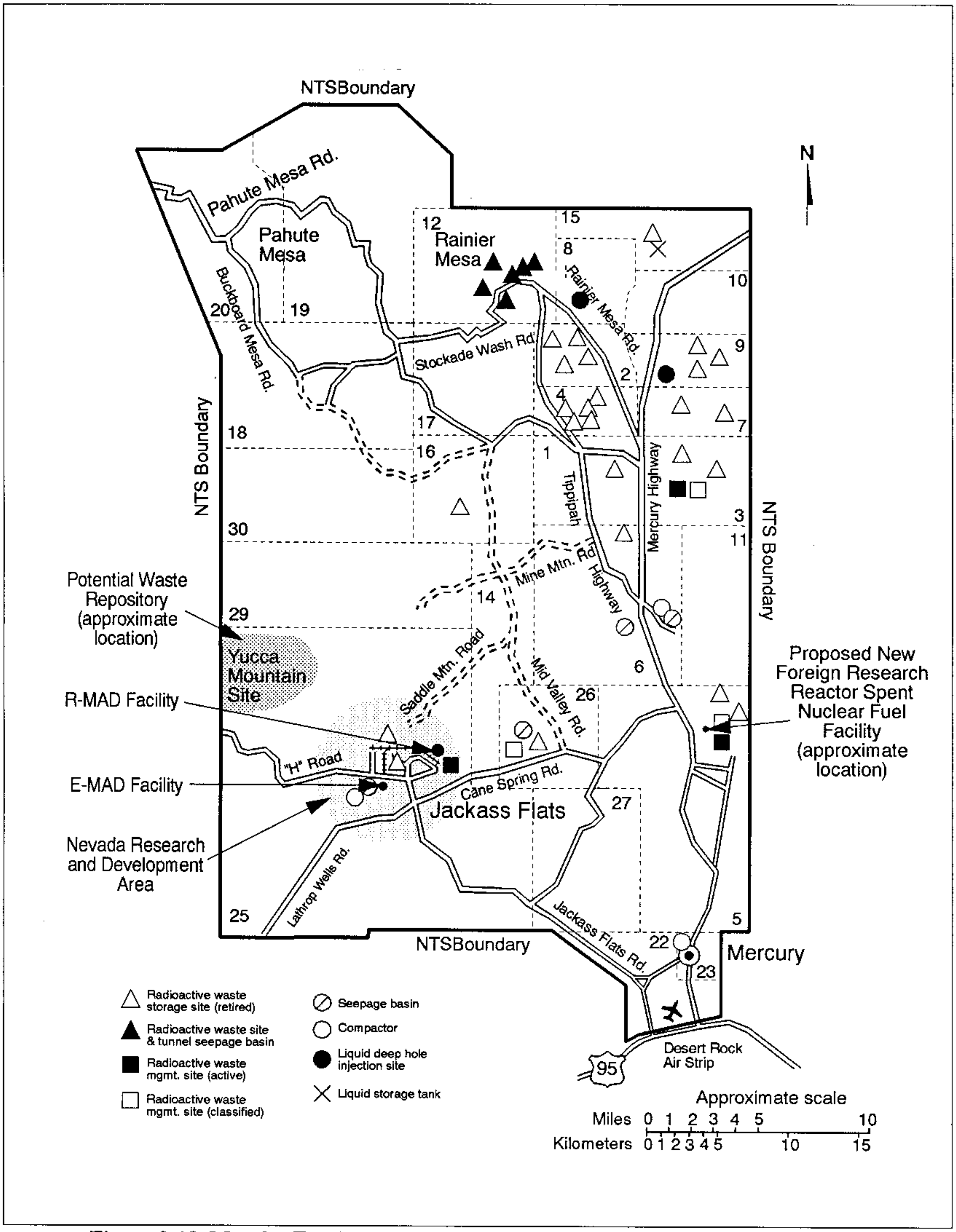


Figure 2-18 Map for Foreign Research Reactor Spent Nuclear Fuel Storage at the Nevada Test Site

Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Nevada Test Site would be bounded by analysis option 5A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis option 5A.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. The various arrangements would affect the amount of spent nuclear fuel that would be managed in the United States as the foreign research reactors would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, considered under analysis option 5A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management options at the Nevada Test Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Nevada Test Site for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 5B, which is similar to 5A, is considered as follows:

5B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes (i.e., during Phase 2) would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel (22,700 elements). If the Nevada Test Site were to receive only TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only western spent nuclear fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Nevada Test Site would not be considered as a site for chemical separation. The Nevada Test Site is also not considered for the Hybrid Alternative discussed in Section 2.4.

Table 2-15 presents an overview of the foreign research reactor spent nuclear fuel management options, quantities of foreign research reactor spent nuclear fuel assumed for the analysis, and facilities considered.

Table 2-15 Proposed Quantities of Foreign Research Reactor Spent Nuclear Fuel and Management Options at the Nevada Test Site

Management Alternative 1		*FRR SNF Elements*	*Percentage of Total FRR SNF Elements*	*Storage Option/Technology*		
				Dry Storage		*Wet Storage*
				E-MAD[a]	*New*	*New*
All FRR SNF	Phase 2[b]	22,700	100%	A	A	A
Western FRR SNF	Phase 2	6,300	28%	A	A	A
TRIGA FRR SNF	Phase 2	4,900	22%	A	A	A

A = Applicable
NA = Not Applicable
FRR = foreign research reactor
SNF = spent nuclear fuel
[a] *E-MAD could be available for use five years after the start of implementation.*
[b] *Phase 2 values represent total number of foreign research reactor spent nuclear fuel elements requiring management at the site.*

2.7 Characteristics of Emergency Management and Response

This section addresses the emergency management and response infrastructure that exists to support the possible implementation of those management alternatives of the proposed action that would have an impact in the United States. This section considers emergency management and response at the ports of entry, along ground transport routes, and at the management sites.

2.7.1 DOE and the National Response System

In the United States, State and local governments are responsible for emergency management and response programs. These programs must be capable of managing all hazards ranging from natural disasters to hazardous material incidents on a day-to-day basis. In order to maintain these programs, various State, Tribal, and local governments receive Federal funding. DOE historically has provided a variety of support to governmental jurisdictions in fulfilling its responsibilities under regulatory and National emergency plan taskings (FEMA, 1994; Rogoff, 1994; and DOE, 1994g).

There are three national emergency response plans (i.e., Federal Response Plan, Federal Radiological Emergency Response Plan, and the National Contingency Plan) under which DOE provides radiological monitoring and assessment assistance. Under these plans, DOE provides technical advice and assistance to State, Tribal, and local agencies involved with a radiological incident (DOE, 1989). For a foreign research reactor spent nuclear fuel incident, DOE actions would be guided by the Federal Radiological Emergency Response Plan (FEMA, 1985) and its own internal emergency management and response system.

DOE maintains an emergency management and response system that is based on regulatory requirements as outlined in various DOE Orders (e.g., DOE Order Series 5500 and 5530). These orders require an emergency management and response system that generally follows the models and practices established by the Federal Emergency Management Agency, National Fire Protection Association, American National Standards Institute, and National Council on Radiation Protection and Measurements.

2.7.2 Foreign Research Reactor Spent Nuclear Fuel Transportation

Foreign research reactor operators, their shipping agents, and commercial carriers would have the primary responsibility to coordinate and arrange all activities associated with foreign research reactor spent nuclear fuel shipments and cask return including emergency management and response. DOE, along with other Federal agencies (e.g., Department of Transportation, NRC, Federal Emergency Management Agency, U.S. Department of Defense, and the U.S. Environmental Protection Agency) would provide support and assistance to State, Tribal, and local government agencies responsible for responding to a foreign research reactor spent nuclear fuel incident.

DOE fulfills its role and responsibilities as the Federal agency tasked with developing and maintaining a capability to safely manage spent nuclear fuel (DOE, 1995c), in part by setting overall spent nuclear fuel program management responsibility and policy for transportation and emergency management and response; resolving policy questions; issuing guidance; providing information; and accomplishing oversight by including regulatory compliance requirements in its spent nuclear fuel related contracts and by monitoring the performance of those involved.

According to DOE records, from 1985 to 1993 there were 102,213 DOE shipments consisting of 1,009,357.6 metric tons (1,112,626 tons) of radioactive material. Of these, 457 shipments, containing 13,176.86 metric tons (14,525 tons) were spent nuclear fuel (these weights include the packaging) (DOE, 1994d). To date, there are no records of radiological fatalities that have occurred in the United States due to transportation accidents. To date, no spent nuclear fuel transportation cask has ever been punctured or released any of its radioactive contents, even in actual highway accidents (NRC, 1993).

2.7.3 External Coordination

Historically, DOE ensures coordination with various organizations and agencies through its interaction with Government, national, and local groups such as the Southern States Energy Board and Western Governors' Association, among others.

The primary responsibility for developing and maintaining a radiological hazardous materials emergency response capability is vested in State, Tribal, and local agencies. DOE, on an "as needed," case-by-case basis, has helped State, Tribal, and local agencies prepare for response to potential accidents involving DOE radioactive material shipments (including spent nuclear fuel). As with the Urgent Relief foreign research reactor spent nuclear fuel transportation effort, DOE has offered various types of technical assistance to the affected jurisdictions (SSEB, 1994).

One example of this partnership with State and local governments occurred in August 1994. In support of Urgent Relief safe transportation, special Radiological Emergency Training for Local Responders and Emergency Response Workshop courses were conducted by DOE for approximately 160 local responders from North Carolina and South Carolina (Analysas Corporation, 1994).

2.7.3.1 Financial and Technical Assistance to States and Tribes

DOE provides funding to States and Tribes through the Office of Environmental Management and the Office of Civilian Radioactive Waste Management to assist with transportation related issues. While some of these funding efforts are not directly related to spent nuclear fuel shipments, they do enhance a jurisdiction's emergency management and response capabilities (DOE, 1994g). Financial assistance to States and Tribes for Transportation programs for fiscal year 1994 is shown in Table 2-16 (DOE, 1994g).

Table 2-16 DOE Summary of Financial Assistance to States and Indian Tribes

Total Allocations for Transportation Programs: FY 1994	
Activity	***Amount***
Waste Isolation Pilot Plant	$1,410,848
Cesium Shipment Support	330,000
Spent Fuel Shipment Support	125,000
Transportation External Coordination Working Group[a]	34,921
Urban Energy & Transportation Corporation[b]	150,137
Office of Civilian Radioactive Waste Management	1,332,000
State of Washington Emergency Management Funds	637,570
Total Allocations	**$4,020,476**

[a] *The amount shown reflects the cost to DOE of furnishing travel, food, and lodging for non-DOE participants at two Transportation External Coordination meetings. Participation in Transportation External Coordination meetings is not restricted to States and Tribes; however, it is not possible to break out State and Tribal costs separately.*

[b] *The amount shown reflects the cost to DOE of furnishing travel, food and lodging for non-DOE participants at three Urban Energy & Transportation Corporation meetings. Urban Energy & Transportation Corporation is a non-profit corporation organized primarily to address local government concerns.*

Besides funding, much of DOE's assistance is provided in the form of technical assistance, for which DOE bears the cost. Assistance may be provided through DOE's Radiological Assistance Program and under the National Contingency Plan, as well as through training, DOE sponsored meetings, informal discussions, and informational materials (DOE, 1994g).

2.7.3.2 Training Assistance to States and Tribes

State, Tribal, local personnel, and other Federal agencies participate in training programs developed by DOE for its staff and contractors. State, Tribal, and local personnel pay their own travel and per diem expenses; however, DOE bears the cost of developing and implementing the training. Available training includes:

- Hazardous Waste Transportation and Packaging Workshop which covers regulations governing transportation of radioactive materials;
- Radiological Emergency Response and Operations is offered in conjunction with the Federal Emergency Management Agency, and teaches response to and management of radiological incidents;
- Radiological Emergency Training for Local Responders was piloted during fiscal year 1994 in Wyoming and brings training directly to the states, allowing them to train in their own environment;
- Radioactive Material Response Orientation provides a 1-day introduction for response personnel;
- Advanced Radioactive Materials Transportation Accident Response is a sequel to the previous course for State, Tribal, Regional, and local emergency responders; and

- Transportation Emergency Training for Response Assistance includes several modules. The Public Affairs module, which is specifically designed to include States and Tribes, is scheduled for piloting during Fiscal Year 1995 (DOE, 1994g).

In addition, the DOE-funded Radiation Emergency Assistance Center/Training Site located in Oak Ridge, Tennessee, conducts courses in medical management of radiation emergencies. These courses include:

- Handling of Radiation Accidents by Emergency Personnel;
- Medical Planning and Care in Radiation Accidents;
- Health Physics in Radiation Accidents; and
- Occupational Health in Nuclear Facilities (REACT/TS, n.d.a.).

2.7.3.3 Transportation External Coordination/Working Group

DOE recognizes the need for ongoing partnerships with external organizations: health and safety; emergency management and response; law enforcement; technical; State, Tribal, and local government; and industrial organizations involved in radiological emergency response. This "stakeholder" involvement has been formalized in the Transportation External Coordination/Working Group.

The Transportation External Coordination/Working Group (Table 2-17) is a 35 member body of emergency management and response professional associations (DOE, 1994i). Through this group DOE looks at crosscutting transportation and emergency response issues that all DOE programs either are addressing or will address in the future. In turn, these groups are able to provide input to DOE for its decision-making process involving these issues (Holm, 1994).

2.7.3.4 Transportation Emergency Preparedness Program

The Transportation Emergency Preparedness Program helps integrate DOE's existing emergency management and response capabilities into an effective response system for transportation incidents involving DOE shipments. Through its extensive external coordination program with State, Tribal, and local agencies, DOE develops interfaces for meeting its various national response plan taskings to provide radiological monitoring and assessment technical assistance needed for transportation incidents involving radioactive materials including any possible incidents associated with a foreign research reactor spent nuclear fuel shipment.

Under the Transportation Emergency Preparedness Program Field Assistance Program, DOE provides support for emergency exercises (Table 2-18) that include State, Tribal, and local agencies through the Operations Offices (DOE, 1994g; DOE, 1994n; SSEB, 1994).

2.7.3.5 Radiological Assistance Program

The primary DOE response groups that would assist at a foreign research reactor spent nuclear fuel incident are the Radiological Assistance Program teams that operate from eight strategically located DOE Regional Coordinating Offices (Figure 2-19) around the country. These teams, upon State, Tribal, or local jurisdiction request, provide technical expertise and assistance to monitor and assess radiological hazards. Figure 2-19 displays pertinent information for contacting each regional office.

Table 2-17 Transportation External Coordination/Working Group Membership

Invited Organizations	*Guest Organizations*[c]
American Association of State Highway and Transportation Officials[a]	Emergency Services Representatives
American College of Emergency Physicians	Arizona Division of Emergency Services
AFL-CIO Transportation Trades Department[a]	City of Jacksonville, FL, Fire Department
American Indian Law Center[a]	Louisiana State Police/TESS
American Trucking Association[a]	St. Charles Parish Department of Emergency Management
Association of American Railroads	Ohio Emergency Management Agency
Chemical Manufacturers Association[b]	County Representatives
Columbia River Inter-tribal Fish Commission	Transportation Advisor, Nuclear Waste Project, Carson City, NV
Commercial Vehicle Safety Alliance	Clark County Comprehensive Planning Department, Esmeralda, NV
Conference of Radiation Control Program Directors	Transportation Planner, Nuclear Waste Division, Las Vegas, NV
Cooperative Hazardous Materials Enforcement Development[a]	Mineral County Office of Nuclear Projects, Hawthorne, NV
Council of Energy Resource Tribes	Nye County, NV
Council of State Governments, Midwestern Office	Nye County Nuclear Waste Repository Office, Tonopah, NV
Edison Electric Institute	White Pine County, NV
Emergency Nurses Association	White Pine County Nuclear Waste Project Office, Ely, NV
Hazardous Materials Advisory Council[a]	Tribal Representatives
International Association of Chiefs of Police	Manager, ERWM Program, Nez Perce Tribe, Lapwai, ID
International Association of Fire Chiefs	Industry Representatives
International Association of Fire Fighters	Union Pacific Railroad
International City Management Association[a]	Environmental Evaluation Group
National Association of Counties	PIC
National Association of Emergency Medical Technicians	
National Association of Regulatory Utility Commissioners[a]	
National Conference of State Legislators	
National Conference of State Transportation Specialists[a]	
National Congress of American Indians	
National Coordinating Council on Emergency Management	
National Emergency Management Association	
National Governors' Association[a]	
National Indian Policy Center	
National League of Cities[a]	
Southern States Energy Board	
Urban Energy and Transportation Corporation	
Western Governors' Association	
Western Interstate Energy Board	

[a] *Denotes organizations invited to attend, but have not yet participated.*

[b] *Denotes organizations invited to attend, but do not wish to participate.*

[c] *Denotes organizations who have attended Transportation External Coordination/Working Group meetings, but are not full-time members.*

Table 2-18 Radiological Emergency Response Exercises

Exercise	Location	Date
TRANSAX 1994	Ontario, OR	August 3, 1994
TRANSAX 1993	Lamy, NM	September 1, 1993
TRANSAX 1992	Fort Hall, ID	September 16, 1992
TRANSAX 1990	Colorado Springs, CO	November 8, 1990
WIPPTREX 93-1	Laramie, WY	April 14, 1993
WIPPTREX 92-1	Raton, NM	October 28, 1992

Radiological Assistance Program teams are composed of a range of technical specialists who volunteer for team membership. The teams are activated on an "as needed" basis and generally can be ready to deploy from their home station within 4 hours of notification. Response times to the scene vary depending on the accident's location and the level of assistance required (Taylor, 1995).

In 1994, Radiological Assistance Program teams responded 37 times to a variety of radiological situations throughout the country. Upon evaluation, a number of these responses were determined to be nonradiological hazards (Taylor and Hauptman, 1994).

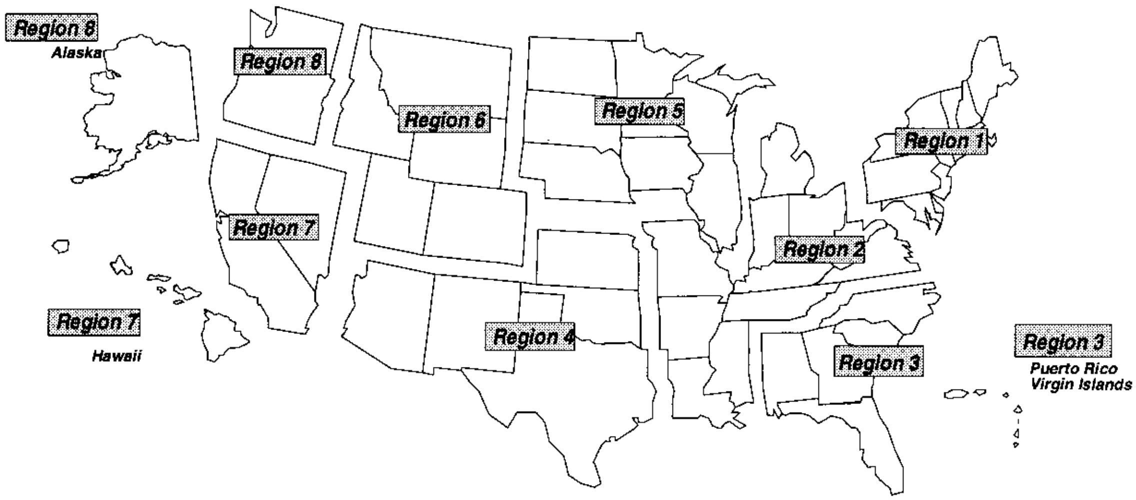
Typically, Radiological Assistance Program teams are involved in identifying personnel, equipment, or property that may be radiologically contaminated, recommending sources of medical advice for the treatment of personal injuries sustained as a result of exposure to radioactivity, and providing advice or assistance in monitoring, decontamination, material recovery, or other post-emergency operations (Gordon-Hagerty, 1993).

2.7.4 Emergency Management and Response at Ports of Entry

From 1979 to 1992, 317 spent nuclear fuel shipments in "Type B" transportation casks were made through various United States ports (NRC, 1993) with no releases of radioactive materials. The "Type B" cask shipments were placed in standard maritime shipping containers the same as any other material being sent to the port. Foreign research reactor spent nuclear fuel shipments are subject to the same types of potential hazards as those of other ships carrying nonradioactive hazardous materials.

Under the Oil Spill Prevention Act of 1990, each port is required to develop an Area Contingency Plan. While the main focus of these plans is an oil spill response, they have been expanded in many cases to address other types of hazardous material responses including radioactive material. These plans outline response capabilities, procedures, and authorities for responding to and recovering from hazardous material incidents.

These ports of entry have a specially designated and prepared terminal or dock area for unloading hazardous materials. They have either a dedicated hazardous material response team or access to a local team through some type of mutual aid agreement. These emergency response teams receive ongoing training and participate in various types of drills and exercises. Also, the dock workers receive varying levels of ongoing hazardous materials response training.



Regional Coordinating Office	Mailing Address	Telephone for Assistance
1. Brookhaven National Laboratory	Upton, Long Island New York 11973	(516) 282-2200 24 Hours
2. Oak Ridge Operations Office	DOE EOC M/S DP 801, P.O. Box 2001 200 Administration Road Oak Ridge, TN 37831-8543	(615) 576-1005 24 Hours
3. Savannah River Operations Office	P.O. Box 616 Aiken, SC 29808	(803) 725-3333 24 Hours
4. Albuquerque Operations Office	P.O. Box 5400 Albuquerque, NM 87185-5400	(505) 845-4667 24 Hours
5. Chicago Operations Office	9800 S. Cass Avenue Argonne, IL 60439	Duty Hours: (708) 252-4800 Off Hours: (708) 252-5731
6. Idaho Operations Office	850 Energy Drive Idaho Falls, ID 83401-1563	(208) 526-1515 24 Hours
7. Oakland Operations Office	1301 Clay Street Oakland, CA 94612	(510) 637-1794 24 Hours
8. Richland Operations Office	825 Jadwin Avenue Richland, WA 99352	(509) 373-3800 24 Hours
9. DOE Headquarters EOC	1000 Independence Avenue Washington, DC 20585	Non-Emergency: (202) 586-8100 Emergency: (202) 586-8500

Figure 2-19 DOE Regional Coordinating Offices for Radiological Assistance and Their Geographical Areas of Responsibility

The U.S. Coast Guard Captain of the Port would have primary On-Scene Coordinator responsibility during a foreign research reactor spent nuclear fuel incident/accident at the port, and would work in conjunction with emergency responders from the Port Authority and local jurisdictions. The On-Scene Coordinator also would be able to call upon a wide range of U.S. Coast Guard resources and the resources of other Federal agencies.

On-Scene Coordinators have at their disposal the resources of the staff of the Marine Safety Office, the resources of the staff at U.S. Coast Guard Headquarters in Washington, DC, assets of any Boat Stations in the Marine Safety Office zone, and any Air Groups. The Marine Safety Office or a Port Authority facility is used as the Emergency Operations Center for many incidents. The On-Scene Coordinator has the authority to call in the Strike Team.

These Strike Teams have limited responsibilities in the course of an incident. Their two main duties are containment and clean-up. They use booms, skimmers, absorbents, and chemicals in their response. The vessels used as platforms for booms and skimmers are usually provided by the Marine Safety Office or Boat Station in the area of the incident. Strike Teams consist of highly trained pollution response and clean-up personnel.

There are three Strike Teams under the command of the National Strike Force Coordination Center in Elizabeth City, NC, that could be called on for a foreign research reactor spent nuclear fuel accident. These teams are located on the three coasts of the United States: the Gulf Strike Team located in Mobile, AL; the Pacific Strike Team located in Novato, CA; and the Atlantic Strike Team located in Fort Dix, NJ.

For a foreign research reactor spent nuclear fuel accident, the U.S. Coast Guard On-Scene Coordinator would request an NRC or DOE representative. DOE Radiological Assistance Program teams would be requested as needed.

2.7.5 Emergency Management and Response Along Ground Transport Routes

During transport of the spent nuclear fuel received as a result of the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel Environmental Assessment, the foreign research reactor operator's shipping agents were required to ensure that all activities of the agent's and the commercial carrier's personnel conformed to regulatory requirements, as well as all plans and procedures developed for the foreign research reactor spent nuclear fuel (DOE, 1994m). DOE and other Federal and State government agencies monitored the activities of the shipping agent and the commercial carrier to ensure that the regulatory requirements were met. If foreign research reactor spent nuclear fuel is managed in the United States, DOE will prepare a Transportation Plan before any shipments are undertaken. The Transportation Plan will detail all transportation activities necessary for the safe and secure transport of the foreign research reactor spent nuclear fuel from the point of origin to the management site in the United States. The general provisions for such a plan are included in Appendix H.

Primary responsibility for emergency response to a foreign research reactor spent nuclear fuel incident would reside with local authorities (DOE, 1989). Each corridor State or Tribe would be responsible for augmenting their existing emergency management and response plans and procedures with any foreign research reactor spent nuclear fuel specific information they felt was necessary.

States coordinate with their local jurisdictions on emergency planning and information. States and Tribes would be responsible for notifying DOE of any conditions that could affect the safe and secure transport of foreign research reactor spent nuclear fuel shipments through their jurisdictions. DOE would provide technical advice and assistance to the shippers and affected government jurisdictions to ensure safe transportation.

Spent nuclear fuel shipments transported by rail, barge, or commercial truck carrier would be subject to the same potential problems as any other hazardous materials shipment that travels daily by these means, namely severe weather, mechanical problems, derailments, and collisions.

DOE would seek to mitigate potential highway foreign research reactor spent nuclear fuel accident consequences by ensuring commercial carriers comply with NRC guidelines for shipment security and the Department of Transportation Highway Route Controlled Quantity routing regulations (DOE, 1995c) which are designed to reduce radiological transportation risk impacts.

The rail and barge industries are similar to the trucking industry in the hazardous material transportation regulatory regime. Documentation, manifesting, placarding, labeling, and other communications are controlled by 49 CFR Parts 100-199.

The carriers used for the transportation of foreign research reactor spent nuclear fuel would be required to develop emergency response plans. In developing these plans, the carriers would be required to consider the following responsibilities:

- protect life, health, and the environment;
- notify appropriate railroad officials in a timely manner;
- notify the appropriate Federal, State, and local authorities, and the shipper;
- initiate a prompt and proper response;
- provide appropriate resources and expertise for resolution of the incident;
- perform cleanup functions; and
- establish and maintain a working contact with the responsible Governmental authorities until they declare the incident closed.

As discussed in Section 2.7.2, spent nuclear fuel shipments, like other hazardous material shipments, have been involved in transportation accidents. Those that have occurred have not resulted in a radiological hazard or damage to the public or the environment. This is primarily due to the rigorous packaging.

Each State and Tribe along a shipping route would be notified of the foreign research reactor spent nuclear fuel shipment's itinerary through that jurisdiction to enable the appropriate agencies to notify the necessary response personnel. Also, DOE would maintain continuous communications through its communications and tracking systems (DOE, 1989).

Foreign research reactor spent nuclear fuel shipments would be tracked either by the commercial carrier or by a satellite tracking system similar to DOE's Transportation Tracking and Communications System (Figure 2-20). The satellite tracking system would provide a "real-time" satellite tracking and voice communications system that would link the truck or train and its escorts with a control center. Some commercial carriers have established their own satellite tracking systems. The DOE system would interface with these systems and co-monitor the shipment's progress to ensure maximum accountability and security. The satellite tracking system would also coordinate "SAFE PARKING" requests from the states.

If a situation would arise (e.g., severe weather, mechanical difficulties, protesters, security threat, personnel illness or injury) that presented a hazard or threat to a highway foreign research reactor spent nuclear fuel shipment, DOE would have arranged through Memoranda of Agreement for the commercial carrier to divert to any Federal installation (e.g., a DOE site or military base) and request "SAFE PARKING" at that facility until the situation is resolved. The receiving facility would assist in providing security and logistical support until the shipment was prepared to depart.

State, Tribal, and local agencies, as well as the commercial carriers, maintain various emergency response plans and procedures. During a foreign research reactor spent nuclear fuel highway, barge, or rail accident, the personnel accompanying the shipment would be the immediate contact for information to the local emergency responders having jurisdiction and Incident Commander authority over the situation.

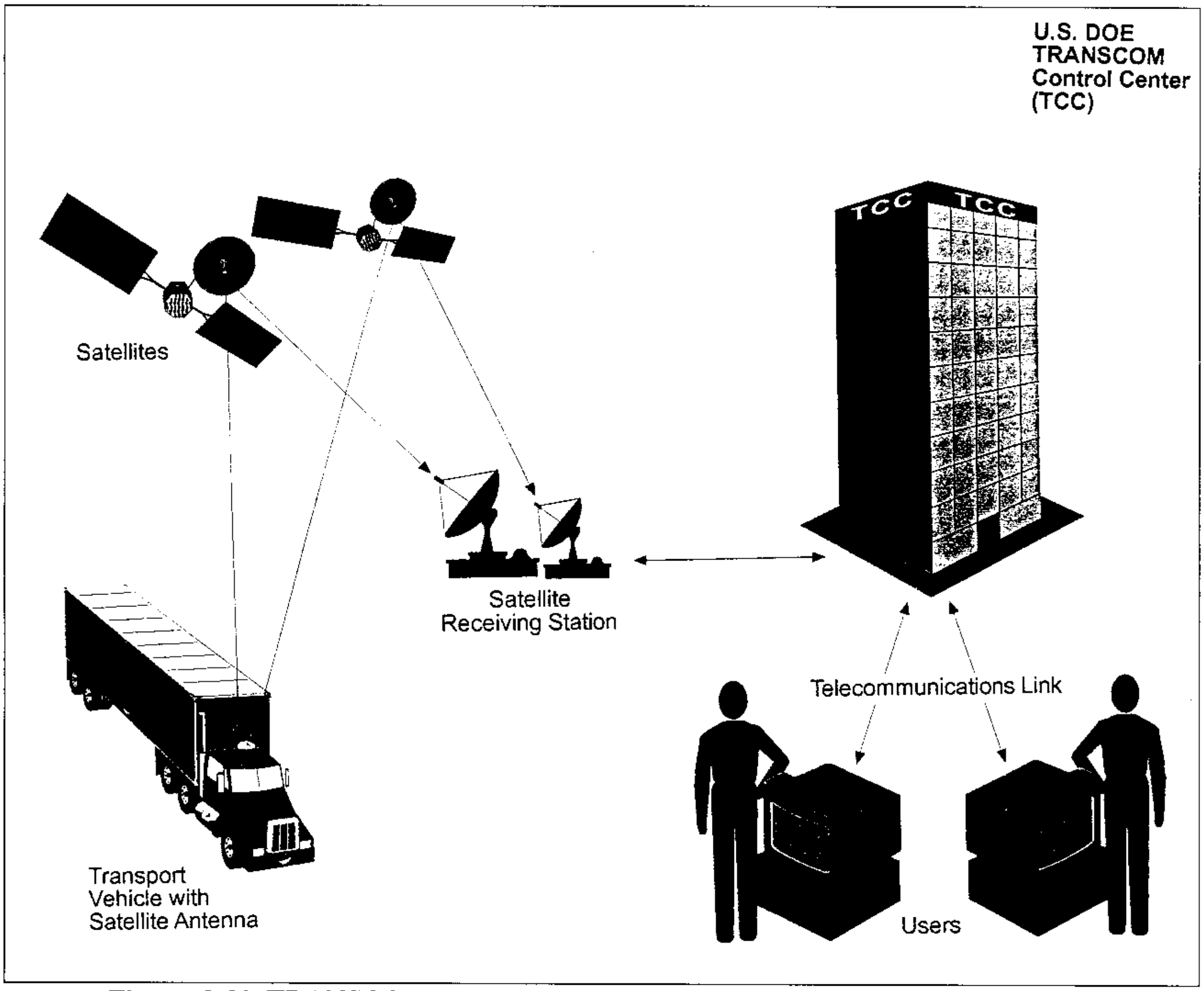


Figure 2-20 TRANSCOM, DOE's Transportation Tracking and Communications System

Additionally, the Hazardous Material Regulations (49 CFR 177.861) advise highway shippers, carriers, and emergency responders to contact DOE if assistance with radioactive materials is required (DOE, 1990b). A DOE Radiological Assistance Program team could respond to the scene if requested.

Incident Commanders have other sources of technical assistance they could call on such as the commercial carrier's technical experts (through a 24-hr contact number), the National Response Center, and the Chemical Transportation Emergency Center, which provides immediate response advice and information from the shipper on a 24-hr basis.

2.7.6 Emergency Management and Response at Management Sites

The DOE 5500 series of Orders incorporates various Federal regulatory requirements and mandates an extensive emergency management and response program at each management site in the same manner as any other industrial facility or local government jurisdiction. State-level specific requirements are addressed by each respective site.

Each of these sites routinely handles hazardous materials that have potential emergency management and response considerations similar to foreign research reactor spent nuclear fuel. These, along with the usual risks posed by any industrial environment, are regularly evaluated through various Safety Analysis Reports and Hazards Assessment studies. These situations are then mitigated to the greatest extent possible.

2.8 Security Measures

Domestic transportation of foreign research reactor spent nuclear fuel would be under the regulatory jurisdiction of the U.S. Department of Transportation and NRC. In the event that foreign research reactor spent nuclear fuel were to be transported through a military port of entry, applicable requirements would be established in advance by the U.S. Department of Defense, DOE and NRC to provide the appropriate level of security.

The objective of the security measures during transportation of spent nuclear fuel are to minimize the possibilities for radiological sabotage of spent nuclear fuel shipments and facilitate the location and recovery of spent nuclear fuel shipments that may have come under control of unauthorized persons. The elements of the security measures would be considered when developing the Transportation Plan to be developed by DOE in consultation with State, local, and Tribal officials prior to any actual spent nuclear fuel shipments. The general provisions of the Transportation Plan, which would include requirements relative to emergency response planning, security considerations, and communications during actual shipments of foreign research reactor spent nuclear fuel, are included in Appendix H.

The security measures provided by the regulations would make the hijacking of a transportation cask a highly unlikely event. In the first place, the large size and weight of these casks (9.1 to 22.7 metric tons, or 10 to 25 tons) and the inherent radioactivity of the spent nuclear fuel make spent fuel in a transportation cask an unlikely hijacking target. For a malicious act of sabotage, there are, in fact, more accessible targets than spent nuclear fuel, that would provide more spectacular detrimental effects; especially considering the fact that, aside from the radioactivity of the spent nuclear fuel, which is a relatively short range effect, the spent nuclear fuel elements are simply pieces of metal (which might be somewhat warm). In the event of a hijack attempt aimed at some long-term use of the contents of the cask, the communications systems required to be used during the shipment would enable timely notification of authorities who would mobilize response forces. Tracking systems would allow the location of the cask to be determined in real time, thereby aiding in the timely interception of the hijackers by response forces.

The successful completion of attempts aimed at short-term destructive acts, such as explosions from within the cask or inducement of criticality, are not considered credible because they would require sufficient time to breach the cask at a great personal risk to the hijackers (probably lethal exposure), special tooling and techniques, and/or the use of specialized materials (for sufficient moderation) that in themselves are safeguarded materials.

Malicious attack scenarios from a distance, such as the explosion of a bomb near a transportation cask, or an attack by an armor-piercing weapon could be within the realm of possibility. The risk to the health and safety of the public associated with such an event cannot be calculated since there is no basis for estimating either the probability of such an event occurring or that damage sufficient to release radioactive material from the cask would occur. Appendix D, Section D.5.9, provides a discussion of the consequences of some sabotage/terrorist initiated events for the purpose of emergency response planning.

2.9 Preferred Alternative

In selecting a preferred alternative for the management of foreign research reactor spent nuclear fuel, DOE and the Department of State took several factors into consideration, including the following:

1. U.S. Government nuclear weapons nonproliferation policies and objectives;
2. DOE responsibilities (e.g., safe handling of hazardous materials, safety/health risks to workers, compatibility with other ongoing missions, etc.);
3. Potential environmental impacts (e.g., public safety, etc.);
4. Public comments received and concerns expressed following issuance of the Draft EIS;
5. Analysis of impacts and alternatives in the Programmatic SNF&INEL Final EIS (DOE, 1995c), as well as the Record of Decision for that EIS;
6. Estimated costs of alternatives for management of foreign research reactor spent nuclear fuel;
7. Public issues/concerns/perceptions (e.g., fairness/equity to affected States and populations, etc.); and
8. Uncertainties (e.g., future budget priorities and continuity of funding, technology development, repository timing and waste form acceptance criteria, regulatory change, etc.).

Based on consideration of these factors, DOE and the Department of State, in consultation with other Government agencies, designate the alternative described below as the preferred alternative. This preferred alternative is the same as Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States, discussed in Section 2.2), with the modifications discussed below. The basic components of Management Alternative 1 have been modified to incorporate various implementation alternatives discussed in Section 2.2.2.

The amount of foreign research reactor spent nuclear fuel that would be accepted and managed, as specified in Section 2.2.1.3, could total approximately 19.2 MTHM, with a volume of approximately 110 m^3 (4,100 ft^3), representing approximately 22,700 individual spent nuclear fuel elements. The target material that would be accepted and managed, as specified in Section 2.2.2.1, contains an additional 0.6 MTHM representing the uranium content of approximately 620 additional typical foreign research reactor spent nuclear fuel elements. The following stipulations on qualifying spent nuclear fuel types would apply:

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors operating on LEU fuel or in the process of converting to LEU fuel when the policy becomes effective.
- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors which operate on HEU fuel when the policy becomes effective and which agree to convert to LEU fuel. Spent nuclear fuel would not be accepted from research reactors that could convert to LEU fuel but refuse to do so.

- Spent nuclear fuel (HEU) would be accepted from research reactors having lifetime cores, from research reactors planning to shut down by a specific date while the policy is in effect, and from research reactors for which a suitable LEU fuel is not available.

- Spent nuclear fuel (HEU and/or LEU) would be accepted from research reactors that are already shut down.

- Unirradiated fuel (HEU and/or LEU) from eligible research reactors would be accepted as spent nuclear fuel.

- For research reactors with both HEU and LEU spent nuclear fuel available for shipment, LEU spent nuclear fuel would not be accepted until the HEU spent nuclear fuel is exhausted, unless there are extenuating circumstances (e.g., deterioration of one or more LEU elements sufficient to cause a safety problem).

- Spent nuclear fuel (HEU and/or LEU) would not be accepted from new research reactors starting operation after the date of implementation of the policy.

The policy duration under this preferred alternative would be 10 years, beginning on the date when the management policy would become effective, as discussed in Section 2.2.1.1. Shipments of spent nuclear fuel to the United States could be made for a period of 13 years, starting from the effective date of policy implementation, as long as the spent nuclear fuel had already been discharged prior to the beginning of the policy period or is discharged during the policy period.

The aluminum-based foreign research reactor spent nuclear fuel would be managed at the Savannah River Site and the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory, in accordance with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the settlement agreement reached between DOE and the State of Idaho [Public Service Co. of Colorado v. Batt, No. CV 91-0035-S-EJL (D. Id.) and United States v. Batt, No. CV-91-0054-S-EJL (D. Id.)]. Under this preferred alternative, up to approximately 19 MTHM of aluminum-based foreign research reactor spent nuclear fuel (approximately 17,800 elements), representing up to approximately 675 casks, and target material representing up to approximately 140 additional casks would be accepted and managed at the Savannah River Site. Also, up to approximately 1 MTHM of TRIGA foreign research reactor spent nuclear fuel (approximately 4,900 elements), representing up to approximately 162 casks would be accepted and managed at the Idaho National Engineering Laboratory.

The candidate U.S. ports of entry are listed in Section 2.2.1.6 and are described in detail in Section 3. Although all of the ports are acceptable based on the port selection criteria discussed in Appendix D, DOE would prefer to candidate use the military ports in proximity to the spent nuclear fuel management sites (i.e., Charleston NWS and the Concord NWS). Under this preferred alternative, a maximum of 38 casks of TRIGA foreign research reactor spent nuclear fuel (estimated to require about 5 shipments) could be accepted at a Western port, with 150 to 300 shipments being accepted via an Eastern port.

The foreign research reactor spent nuclear fuel and target material would be shipped by either chartered or regularly scheduled commercial ships from the foreign ports to the United States, as specified in Section 2.2.1.5.

DOE would take title to the foreign research reactor spent nuclear fuel and target material that is shipped by sea after it is offloaded at the port of entry, and to the spent nuclear fuel and target material shipped solely overland (i.e., from Canada) at the border crossing between Canada and the United States.

The foreign research reactor spent nuclear fuel and target material would be transported from the United States ports to the management sites by truck and rail as specified in Section 2.2.1.7.

The financing arrangement under this preferred alternative would be for the United States to bear the full cost for transporting and managing the foreign research reactor spent nuclear fuel and target material accepted from countries with other-than-high-income economies, and to charge high-income economy countries a competitive fee. The fee would be established in a Federal Register Notice (as opposed to being published in this Final EIS), to allow DOE flexibility to adjust the fee to account for inflation, or changes in spent nuclear fuel management practices in the United States.

For the aluminum-based foreign research reactor spent nuclear fuel, a three point strategy is proposed, as follows:

1. DOE would embark immediately on an accelerated program at the Savannah River Site to identify, develop, and demonstrate one or more non-reprocessing, cost-effective treatment and/or packaging technologies to address potential health and safety issues that may develop and to prepare the foreign research reactor spent nuclear fuel for ultimate disposal. The purpose of any new facilities that might be constructed to implement these technologies would be to change the foreign research reactor spent nuclear fuel into a form that is suitable for geologic disposal, without necessarily separating the fissile materials, while meeting or exceeding all applicable safety and environmental requirements. Examples of technologies that would be considered include: *can-in-canister, chop and dilute/poison, melt and dilute/poison, plasma arc treatment, electrometallurgical treatment, glass material oxidation and dissolution, chloride volatility, dissolve and vitrify, direct disposal in small packages*, etc. Functional schematics of these technologies are shown in Figure 2-21. In conjunction with the examination of new technologies, variations of conventional direct disposal methods would also be explored. After treatment and/or packaging, the foreign research reactor spent nuclear fuel would be managed on site in "road ready" dry storage until transported off-site for continued storage or disposal. DOE would select, develop, and implement, if possible, one or more of these treatment and/or packaging technologies by the year 2000. DOE is committed to avoiding indefinite storage of this spent nuclear fuel in a form that is unsuitable for disposal.

2. Despite DOE's best efforts, it is possible that a new treatment and/or packaging technology may not be ready for implementation by the year 2000. It may become necessary, therefore, for DOE to use the F-Canyon to reprocess some foreign research reactor spent nuclear fuel elements, while the F-Canyon is operating to stabilize at-risk materials as recommended by the Defense Nuclear Facilities Safety Board. (For example, under current schedules this activity could take place between the years 2000 and 2002.) In that event, the foreign research reactor spent nuclear fuel would be converted into LEU and wastes generated during reprocessing. Certain wastes would be vitrified in the Defense Waste Processing Facility, while others would be solidified in the Saltstone facility. In order to provide a sound policy basis for making a determination on whether and how to utilize the F-Canyon for processing tasks that are not driven by health and safety considerations, DOE will commission or conduct an independent study of the nonproliferation and other (e.g., cost and timing) implications of reprocessing spent nuclear fuel from foreign research reactors. The study will be initiated in mid-1996 and will be completed in a timely fashion to allow a subsequent decision about possible use of the F-Canyon for foreign research reactor spent nuclear fuel reprocessing to be fully considered by the public, the Congress and the Executive Branch agencies. Pending disposition of the foreign research reactor spent

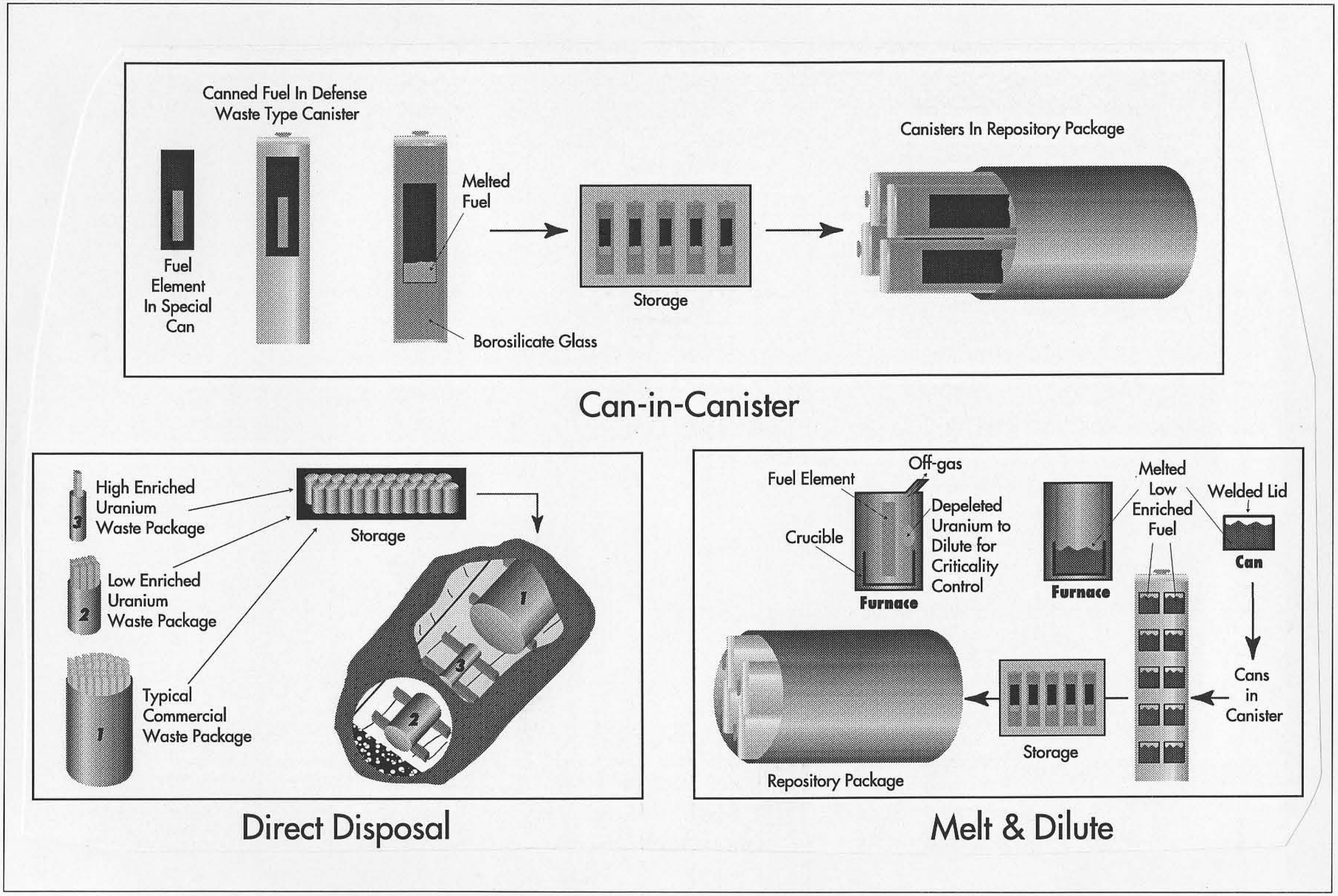


Figure 2-21 New Treatment and Packaging Technologies (Functional Schematic Diagrams)

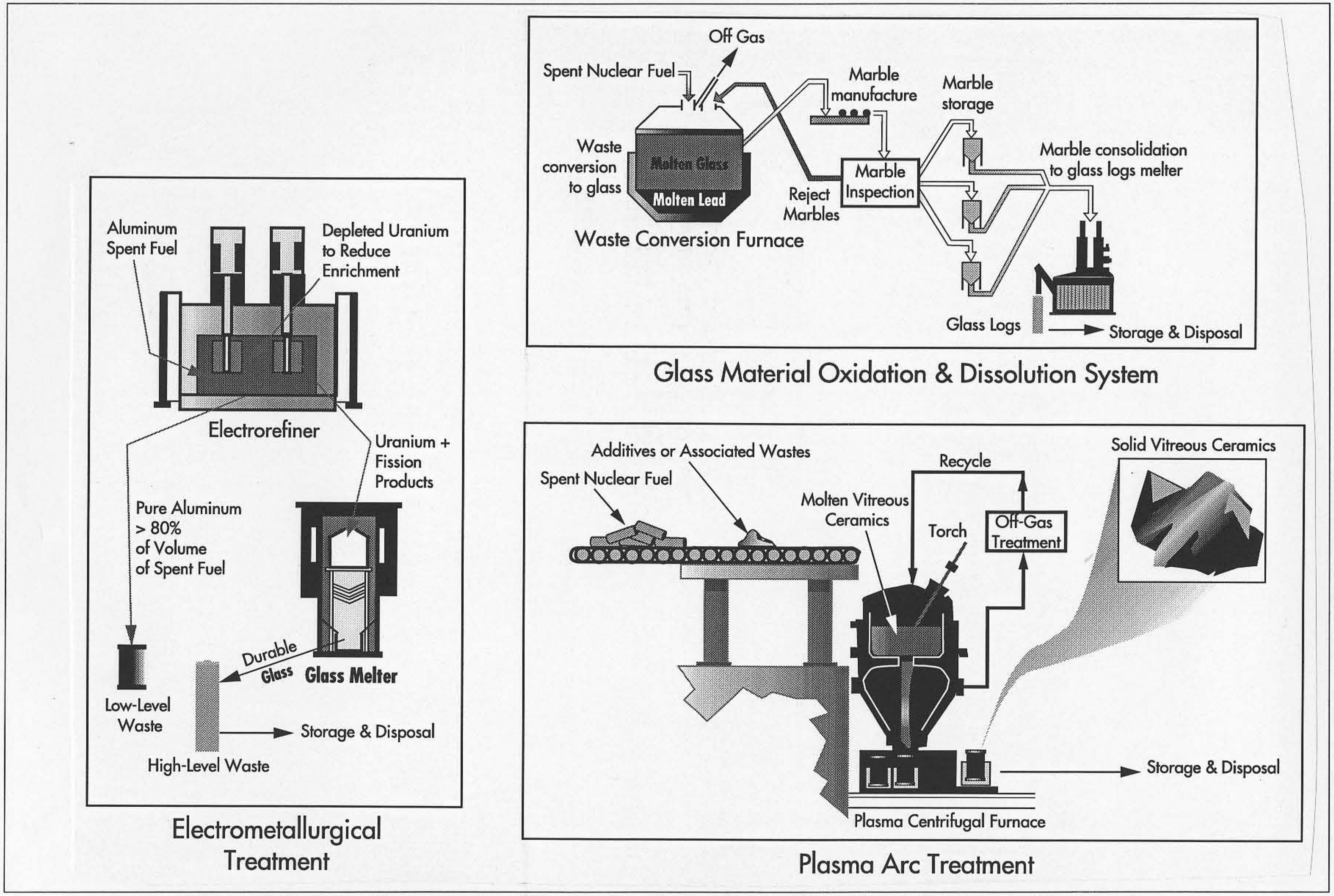


Figure 2-21 New Treatment and Packaging Technologies (Functional Schematic Diagrams) (Continued)

nuclear fuel by either a new treatment and/or packaging technology or reprocessing in the F-Canyon, the spent nuclear fuel would be placed in existing wet storage at the Savannah River Site.

3. DOE would conduct a program of close monitoring of any foreign research reactor spent nuclear fuel and target material that would be accepted for storage in existing wet storage facilities. DOE is presently unaware of any technical basis for believing that this spent nuclear fuel cannot be safely stored until one or more of the treatment and/or packaging technologies becomes available. Nevertheless, if health and safety concerns involving any of the foreign research reactor spent nuclear fuel elements are identified prior to development of an appropriate treatment and/or packaging technology, DOE would use the F-Canyon to reprocess the affected spent nuclear fuel elements, if it is still operating to stabilize at-risk materials.

   Because of criticality constraints stemming from the configuration of the F-Canyon, under no circumstances would it be possible to produce separated HEU that is suitable for a nuclear weapon. Instead, depleted uranium would be added to the foreign research reactor spent nuclear fuel near the beginning of the reprocessing process, so that only LEU would be produced when the uranium is separated from the fission products. The trace quantities of plutonium in the spent nuclear fuel would be left in and solidified along with the high-level radioactive reprocessing wastes. This would further the President's policy to discourage the accumulation of excess weapons grade fissile materials, to strengthen controls and constraints on these materials and, over time, to reduce worldwide stocks.

The TRIGA foreign research reactor spent nuclear fuel would be stored at the Idaho National Engineering Laboratory in the Fluorinel Dissolution and Fuel Storage (FAST) facility (wet storage) or preferably the dry storage Irradiated Fuel Storage Facility (IFSF) and the CPP-749 dry storage area. After 2003, all foreign research reactor spent nuclear fuel would be managed in accordance with the provisions of the settlement agreement between DOE and the State of Idaho, until transported off-site for ultimate disposition. Depending on the nature of any new treatment and/or packaging technology that might be developed, the TRIGA spent nuclear fuel would also be processed using such a new technology, if necessary for disposal.

A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, whose goal is minimizing and eventually eliminating the use of HEU in civil nuclear programs, by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, are dependent on the United States' commitment to action such as that embodied in this preferred alternative.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations and persons as a signal of endorsement of the use of reprocessing as a routine method of waste management for spent nuclear fuel or a reversal of U.S. policy on reprocessing. This would not be an accurate interpretation. The U.S. policy regarding reprocessing was established in Presidential Decision Directive 13. DOE and the Department of State have determined that this preferred alternative is not inconsistent with that policy. The draft version of this EIS indicated that reprocessing is a non-preferred technology and would not be used unless one or more of a set of specific conditions occurred (see Draft EIS Section 2.2.2.6). This final preferred alternative, which includes reprocessing, establishes a prescribed set of circumstances that would have to be met before reprocessing would be used. The independent study discussed above in point 2 of the strategy for management of aluminum-based spent nuclear fuel will review the policy, technology, cost and schedule implications for reprocessing foreign research reactor spent nuclear fuel to determine whether reprocessing of foreign research reactor spent nuclear fuel is justified for other than health and safety reasons.

Policy considerations and environmental impacts associated with implementation of this preferred alternative are presented in Section 4.7. Cost considerations are included in Section 4.9.

Basis for the Preferred Alternative - The elements of the preferred alternative discussed above have been selected based on the following considerations:

1. ***Management Alternative*** - The various management alternatives considered are discussed in Sections 2.2 through 2.4 of the EIS. The analyses in Sections 4.2 through 4.5 of the EIS demonstrate that the impacts on the environment, involved workers, or the citizens of the United States from implementation of any of the management alternatives or implementation alternatives analyzed (other than beneficial impacts associated with support for United States nuclear weapons nonproliferation policy) would be small and completely within the applicable regulatory limits, and would not provide a basis for discrimination among the alternatives. As a result, the process for selection of the elements of the preferred alternative focused on programmatic considerations:

   a. DOE and the Department of State concluded that the No Action Alternative and Management Alternative 2, Implementation Alternative 1a (Overseas Storage) would be unacceptable since these alternatives are not consistent with United States nuclear weapons nonproliferation policy objectives.

   b. DOE and the Department of State believe that the basic implementation of Management Alternative 1 would be undesirable to the extent that it would involve indefinite storage of foreign research reactor spent nuclear fuel in a form that is not suitable for disposal. Management Alternative 1 modified to rely solely on Implementation Alternative 6 (Near Term Conventional Chemical Separation in the United States) would raise nuclear weapons nonproliferation policy questions. Management Alternative 1 modified to rely solely on Implementation Alternative 7 (Developmental Treatment and/or Packaging Technologies) could not be selected at this time because no decision has been made on which technology will be pursued.

   c. DOE and the Department of State also believe that Management Alternative 2, Implementation Alternative 1b (Overseas Reprocessing) would be technically complex and potentially extremely expensive because it would require the United States to accept reprocessing wastes from the overseas reprocessing operations. This is due to the fact that both of the countries in which the overseas reprocessing might be accomplished require the reprocessing wastes to leave their countries, and many of the countries that would be covered by the proposed policy cannot accept the return of such reprocessing wastes. The intermediate-level radioactive wastes produced in Europe during reprocessing of research reactor spent nuclear fuel are often in a concreted waste form, unlike any high-level radioactive waste form in the United States. This concreted waste form has not been evaluated for disposal in a United

States geologic repository. Accordingly, acceptance of such waste in the United States likely could require expensive, currently unproven treatment and/or packaging technologies to transform it into a form that would be acceptable for disposal.

d. The sample hybrid alternative (Management Alternative 3) analyzed in the Draft EIS involved partial reprocessing overseas coupled with partial management in the United States. In order for this alternative to be consistent with United States nuclear weapons nonproliferation policy objectives, certain conditions would have to be met by either the reprocessor (e.g., Dounreay) or the research reactor operators. Staff from both DOE and the Department of State have addressed this issue with representatives of the United Kingdom Department of Trade and Industry and reactor operators, and have determined that it would not be possible to ensure compliance with the United States nuclear weapons nonproliferation policy objectives. The primary concern was the inability to ensure that any separated HEU would be blended down to LEU. Obtaining the reactor operators' agreement to such a policy would likely require significant financial subsidies. The potential cost of achieving agreement to blend down the uranium, plus uncertainties regarding Dounreay's long-term availability, led DOE and the Department of State to conclude that successful implementation of this alternative could not be relied on.

None of the alternatives analyzed in the Draft EIS could be implemented without some degree of difficulty. However, a modification of Management Alternative 1 (Manage Foreign Research Reactor Spent Nuclear Fuel in the United States), incorporating a combination of alternatives to the basic implementation components balances policy, technical, cost and schedule requirements. DOE and the Department of State consider that this approach provides the highest assurance that programmatic requirements could be met. This combination also provides the strongest support for United States nuclear weapons nonproliferation policy objectives as all aspects of the alternative would be under the control of DOE, either directly or through the spent nuclear fuel acceptance contracts with the reactor operators.

2. ***Management Technology*** - The alternative spent nuclear fuel management technologies considered are discussed in Sections 2.2.2.7 and 2.6.5 of the EIS. The approaches fall into four broad categories, as follows:

Wet Storage - Wet storage is a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if it were used to implement the proposed action. Furthermore, DOE currently has wet storage facilities in operation at the Savannah River Site and the Idaho National Engineering Laboratory that could be used for storage of foreign research reactor spent nuclear fuel. However, wet storage requires attention to ensure that the storage conditions do not foster slow degradation of the spent nuclear fuel through corrosion.

Dry Storage - Dry storage is also a proven technology, that would also have no more than small impacts, completely within the applicable regulatory limits, if used to implement the proposed action. It is the storage medium that is being selected at all commercial power reactor sites where additional storage capacity is being built. However, it has not been used for research reactor spent nuclear fuel in the United States. Dry storage capacity could be provided at the management sites in time to meet the program's projected needs, if initial spent nuclear fuel receipts were placed into the available wet storage.

Chemical Separation - Chemical separation is also a proven technology, the impacts of which would be small, and completely within the applicable regulatory limits, if used to implement the proposed action. However, DOE is phasing out its chemical separation activities and is currently reprocessing only at the Savannah River Site to stabilize materials for health and safety reasons. Because these chemical separations facilities could be used to treat the foreign research reactor spent nuclear fuel, they provide a contingency to be considered pending availability of an alternate means of treating and/or packaging the spent nuclear fuel prior to ultimate disposition.

New Technologies - Due to concerns regarding geologic disposal of intact spent fuel containing HEU (i.e., the possibility of uncontrolled criticality incidents), some form of treatment of this spent nuclear fuel may be required. While several concepts have been proposed for new treatment and/or packaging technologies, none of them are ready for implementation at this time. Prior to a decision leading to their implementation, additional development work would be required to determine whether and how they could be implemented, based on technical and cost considerations.

In order to effectively implement the preferred alternative of accepting and managing the foreign research reactor spent nuclear fuel in the United States, DOE and the Department of State developed the three point strategy for management of aluminum-based spent nuclear fuel discussed earlier in this Section. This strategy draws on the strengths of each of the spent nuclear fuel management technologies discussed above, while avoiding sole reliance on any of them. Due to the relatively more robust nature of the TRIGA spent nuclear fuel, DOE believes that minimal additional development may be needed to prepare it for storage and final disposition. Accordingly, the preferred alternative specifies that the TRIGA spent nuclear fuel would be placed in existing dry storage facilities at the Idaho National Engineering Laboratory. However, the program to qualify the final geologic disposal form for the TRIGA spent nuclear fuel will continue and the appropriate treatment, if any, would be identified and implemented.

3. ***Policy Duration*** - The alternative policy durations considered are defined in Sections 2.2.2.1 and 2.2.2.2 of the EIS. Analysis of these alternatives concluded that the 5-year option is likely to provide insufficient time for the reactor operators to arrange for alternative spent nuclear fuel disposal mechanisms, and thus might result in some reactor operators refusing to cooperate fully with United States nuclear weapons nonproliferation programs. This, in turn, could undermine international cooperation with other nuclear weapons nonproliferation programs the United States might seek to implement.

   On the other hand, the analysis determined that there was insufficient benefit to be gained from indefinite acceptance of all of the spent nuclear fuel containing HEU because such an approach likely would provide insufficient incentive for other countries to proceed expeditiously with arrangements for alternative disposal mechanisms not involving the United States.

   The approach incorporated into the preferred alternative allows sufficient incentive to the reactor operators to ensure their cooperation, while specifying a definite cut-off point. This alternative provides sufficient lead time to allow the reactor operators to make other arrangements for disposition of their spent nuclear fuel, and provides sufficient time to accept all spent nuclear fuel containing HEU enriched in the United States.

4. ***Amount of Material to Manage*** - The alternative amounts of material that might be covered by the proposed policy are defined in Sections 2.2.1.3 and 2.2.2.1 of the EIS. DOE and the Department of State concluded that management of spent nuclear fuel only from other-than-high-income economy countries would strongly encourage the resurgence of the use of HEU in the high-income economy countries, as well as opening the United States, fairly or unfairly, to charges that we are not living up to our commitments under the *Treaty on the Non-Proliferation of Nuclear Weapons*. Management of only spent nuclear fuel containing HEU would penalize those reactors that have already converted to the use of LEU fuel, and would provide an incentive for reactors to continue to use HEU fuel, or switch back to its use. The impacts that would result from management of any of these different amounts of material would be small, and within the applicable regulatory limits.

   DOE and the Department of State concluded that management of all of the aluminum-based and TRIGA foreign research reactor spent nuclear fuel currently in storage or projected to be discharged during the policy period, and target material containing uranium enriched in the United States, would provide the best support for the objectives of the proposed policy. Implementation of this preferred alternative would provide an opportunity for removal of the maximum amount of HEU from civil commerce and would provide an incentive for the continued conversion to and use of LEU as fuel for foreign research reactors, in place of highly enriched (weapons grade) uranium.

5. ***Marine Transport*** - The alternative approaches to marine transport of foreign research reactor spent nuclear fuel are discussed in Section 2.2.1.5 of the EIS. The analysis in the EIS demonstrates that the impacts to the environment, workers or the public from transport of the spent nuclear fuel using any of these types of ships would be small, and within the regulatory limits. The analyses do not identify any difference in the small impacts that would result from the use of purpose-built vs. general purpose ships. Since "military transports" are in fact the same type of ship as the chartered commercial cargo ships and are crewed by civilians, use of "military transports" would not actually result in any difference in impacts. DOE and the Department of State believe that use of actual warships would be both unnecessary from a security standpoint and could entail additional risk to the environment and the public, since such ships do not routinely carry cargo.

   The approach selected by DOE and the Department of State for the preferred alternative provides maximum flexibility for marine transport.

6. ***Ground Transport*** - The ground transportation alternatives are defined in Section 2.2.1.7 of the EIS. The analyses in the EIS demonstrate that the impacts to the environment, workers or the public, from any of these modes of ground transport (counting barge as a mode of "ground transport") would be small and within the applicable regulatory limits. Furthermore, the differences in potential impacts between the truck, rail and barge alternatives were not significant.

   Both the truck and rail transportation options have been used successfully to transport foreign research reactor spent nuclear fuel in the past. Truck transport was the predominant mode used for over twenty years, until the old "Off-Site Fuels Policy" lapsed in 1988. Rail was the mode used for both shipments under the *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel* (DOE, 1994m). Since neither of the preferred ports of entry (see item 8 below) can reasonably provide barge transport to either of the proposed management sites, barge transport was dismissed from consideration in the preferred alternative.

By providing for either truck or rail transport, the preferred alternative would build on previous satisfactory experience while providing maximum flexibility for dealing with changes in the transportation process in the future.

7. ***Title Transfer Location*** - The alternative points at which DOE might take title to the spent nuclear fuel and target material are discussed in Sections 2.2.1.4 and 2.2.2.4 of the EIS. The point at which title would be transferred has no effect on the physical processes that would take place, and thus would not have any effect on the impacts on the environment, workers or on the public. The Price-Anderson Act would provide liability protection in the unlikely event of a nuclear accident in the United States, whether or not DOE had taken title to the spent nuclear fuel at the time of such an accident. As a result, DOE and the Department of State concluded that the selection of the title transfer location could be made solely on programmatic considerations.

   Acceptance of title at the foreign research reactor sites could make the United States Government liable for any accident that might occur in the country of origin, or on the high seas. DOE and the Department of State have been unable to identify any advantage to the United States of taking title outside the United States.

   Taking title at the limit of United States territorial waters makes the title transfer depend solely on when the ship enters United States waters, which could be difficult for DOE to control in certain circumstances (e.g., a storm).

   Acceptance of title when the foreign research reactor spent nuclear fuel actually enters the land mass of the United States provides the most certainty for implementation.

   The approach incorporated into the preferred alternative ensures that liability for accidents during the transportation process outside the United States would remain with the reactor operators while reinforcing in the minds of the public that the United States Government would be accountable in the unlikely event of an accident within United States territory.

8. ***Ports of Entry*** - The alternative ports of entry considered are discussed in Sections 2.2.1.6 and 3.2 of the EIS. The analyses in the EIS demonstrate that the impacts on either the environment, workers or the public due to use of any of the potential ports of entry analyzed would be small and within applicable regulatory limits.

   Although any one or all of the ten ports of entry described in Sections 2.2.1.6 and 3.2 of this EIS would be acceptable ports of entry, DOE and the Department of State concluded that foreign research reactor spent nuclear fuel marine shipments to the United States should be made via the military ports (selected from among those analyzed in the EIS and found acceptable) in close proximity to the spent nuclear fuel management sites. DOE would seek to transport multiple casks per ship to keep the total number of shipments as low as possible, as well as to reduce risks. The exact number of shipments that might be made would be determined by several factors that are unknown at this time, such as the times at which the reactor operators need to make shipments over the 13 year shipping period, the geographic distribution of the reactors, and the availability of suitable ships that would stop at the required ports to pick up and drop off the spent nuclear fuel and target material.

Use of military ports would provide additional confidence in the safety of the shipments due to the increased security associated with the military ports. It could also require much of the spent nuclear fuel to be shipped via chartered ships since commercial ships would not have stops scheduled at military ports, increasing the cost of spent nuclear fuel shipping. This additional cost would be borne by the reactor operators for shipments from high-income economy countries, and by the United States for shipments from other-than-high-income economy countries. Additional costs would be kept to a minimum by shipping as many casks as possible on each ship (up to a maximum of 8 per ship).

9. ***Management Sites*** - The question of which sites should be used for management of all of DOE's spent nuclear fuel was addressed in the Programmatic SNF&INEL Final EIS (DOE, 1995c). That EIS included consideration of the potential receipt of the foreign research reactor spent nuclear fuel. The Record of Decision for that EIS, issued on May 30, 1995, specifies that any aluminum-based foreign research reactor spent nuclear fuel accepted in the United States shall be managed at the Savannah River Site; and that the remaining foreign research reactor spent nuclear fuel shall be managed at the Idaho National Engineering Laboratory. The site for management of the target material was left to be decided under this EIS. All of the target material currently in DOE's possession is managed at the Savannah River Site. The approach incorporated into the preferred alternative is in compliance with the decision specified in the Record of Decision for the Programmatic SNF&INEL Final EIS.

   The analyses in the EIS demonstrate that the impacts to either the environment or the public through use of any of the sites for management of the foreign research reactor spent nuclear fuel and target material would be small, and within the applicable regulatory limits.

10. ***Financing Arrangement*** - The alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. The financing arrangement used for the proposed action would have no effect on the physical processes that would take place, and thus would not have any effect on the potential impacts on the environment, or on the public. However, it could affect how many foreign research reactor operators elect to ship spent nuclear fuel to the United States. For instance, if DOE and the Department of State chose to charge a full cost recovery fee to all reactors, many, if not all, of the reactors in other-than-high-income economy countries would not have the financial resources to participate. On the other hand, if the United States subsidized all of the reactors, the United States would bear the full financial burden, even for reactors which can afford to pay their fair share.

    DOE and the Department of State concluded that, to ensure that reactor operators in other-than-high-income economy countries would participate in the program, the United States should subsidize receipt of their spent nuclear fuel. DOE and the Department of State also concluded that DOE should strive to recover as much of the cost of managing the spent nuclear fuel as possible from high-income economy countries. DOE concluded that it would announce the fee in a *Federal Register* notice, so that the fee may be changed from time to time as necessary to reflect inflation or improvements in DOE's knowledge concerning the costs of the activities to be carried out.

    Such an approach would encourage participation by as many other-than-high-income economy countries as possible, would recover as much as possible of the United States' expenses for management of spent nuclear fuel from high-income economy countries without encouraging any of them to resort to reprocessing of their spent nuclear fuel, and would provide a mechanism through which to account for inflation and future definition of program details.

2.10 Additional Alternatives Considered But Dismissed

Besides ocean transport by vessel, carriage by air is the only other mode of transportation from overseas nations to the United States. There are two distinct reasons why the air mode is not a feasible alternative to the sea mode for transportation of foreign research reactor spent nuclear fuel.

First, with the possible exception of small sample quantities, spent nuclear fuel is required to be transported in packagings (casks) weighing several tons. As a general rule, casks would have to be shipped singly by air (i.e., one per airplane) because of their weight. This has made the air alternative so costly as to be prohibitive. As a result, there is no commercial operational experience in the United States with air transport of spent nuclear fuel. No "Standard Operating Procedures" have been written and no intermodal transfer procedures (air-truck or air-rail) have been developed. No agreements have been negotiated regarding airspace overflight of other nations or states. Because the United States has no experience with this type of transportation, no meaningful comparison can be made between air transport and ship transport regarding either incident-free doses to workers and the public or accident risks.

Second, plutonium air transport packaging standards clearly apply to movement by air of any non-exceptional package containing more than 0.005 curies of plutonium (10 CFR 71.88a). The foreign research reactor spent nuclear fuel considered in this EIS is non-excepted and could contain more than 0.005 curies of plutonium per cask. Therefore, the spent nuclear fuel would have to be transported in a cask meeting plutonium air transport packaging standards. Because no spent nuclear fuel transportation cask has been certified to meet plutonium air transport packaging standards, transporting foreign research reactor spent nuclear fuel by air to the United States could not be accomplished in the near term.

The following additional considerations contributed to the dismissal of air transport as an alternative transportation mode of foreign research reactor spent nuclear fuel: 1) Most United States airports lack rail connections; therefore ground transportation would be limited to the use of trucks; 2) airports have no experience in handling spent nuclear fuel and the capabilities of the available handling equipment are marginal and; 3) worker exposure associated with handling activities would be higher because a lack of automation in handling equipment.

The alternative of accepting of foreign research reactor spent nuclear fuel only from countries that present a potential nuclear weapons proliferation risk was considered but dismissed. A major drawback inherent in this alternative is that potential proliferant countries might well object to being publicly identified as such and, on one pretext or another, refuse to cooperate with the United States in the program. Further, this alternative would not address the potential that some countries that are not currently identified as nuclear weapons proliferation threats might become such a threat in the future. To account for acceptance of foreign research reactor spent nuclear fuel from such countries, DOE would have to assume and analyze one or more "hypothetical reactors" to estimate the potential environmental impacts. The public noted its objection to such an approach when DOE proposed to accept 150 foreign research reactor spent nuclear fuel elements from one or more unnamed "hypothetical reactors" in the first draft of the *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel* (DOE, 1994m). Finally, implementation of such an alternative would leave unresolved the spent nuclear fuel disposition concerns of the majority of the countries in which foreign research reactors are operating. These countries would be likely to argue, rightly or wrongly, that the United States was not living up to its obligations under the *Treaty on the Non-Proliferation of Nuclear Weapons* to assist nonnuclear weapon states with the peaceful application of nuclear energy. This would damage the credibility of the United States as a reliable partner in the implementation of international nuclear materials management. In consideration of the above summarized flaws, DOE dismissed this alternative from consideration in this EIS.

As a result of public comments, the possibility of managing foreign research reactor spent nuclear fuel on an isolated island was considered, and dismissed. A new facility on an island could not be ready to receive foreign research reactor spent nuclear fuel for at least five years, necessitating temporary management at another location (Savannah River site or Idaho National Engineering Laboratory) for at least the first half of the policy period. Furthermore, management of spent nuclear fuel on such an island is undesirable from the standpoint of security and safety. Provision of physical security would be much more difficult on a remote island than at a mainland site, due to isolation and the greater challenges of protecting open coastlines. Small isolated islands are subject to a greater frequency of severe weather than occurs in the mainland, and after a severe storm it can be more difficult to restore services than it would be in a mainland area.

3. The Affected Environment

This chapter describes the marine, port, and site environments. Marine environments (Section 3.1) would be potentially impacted by the ocean transport of spent nuclear fuel, and port environments (Section 3.2) would be potentially impacted by the transfer of the casks that would contain the foreign research reactor spent nuclear fuel. The affected environment of the potential DOE management sites for storage is addressed in Section 3.3.

3.1 Marine Environment

The ocean is the principal marine environment potentially impacted by foreign research reactor spent nuclear fuel transport. The scientific study of the ocean is commonly referred to as "oceanography." The discipline of oceanography has been subdivided in terms of the basic physical sciences into geological, chemical, physical, and biological oceanography. The purpose of this section is to provide a basic description of the marine environment. It describes those relevant features that have an influence on the general circulation of the world ocean.

3.1.1 Geological Oceanography

Marine geology or geological oceanography is the study of the character and history of that portion of the earth's surface covered by seawater. The world ocean is geographically divided into five major regions: (1) the Southern Ocean, (2) the Atlantic Ocean, (3) the Pacific Ocean, (4) the Indian Ocean, and (5) the Arctic Ocean. The Pacific Ocean occupies roughly 46 percent of the total world ocean area, the Atlantic Ocean approximately 23 percent, the Indian Ocean nearly 20 percent, and the remaining oceans, 11 percent.

The structural features of the ocean basin surface (Figure 3-1) can be divided into five major entities: (1) shore, (2) continental shelf, (3) continental slope and rise, (4) basin (or abyssal plain), and (5) mid-oceanic ridges. The shore region is commonly referred to as that portion of the land mass that has been modified by oceanic processes. The beach is the seaward limit of the shore, and represents a region that is in dynamic equilibrium between the high and low water marks. Extending seaward from the beach face is the continental shelf. It is characterized by a gentle slope of approximately 1:500. The shelf region has an average width of approximately 65 km (40.4 mi), and a water depth of roughly 130 m (426 ft) at the seaward end of the shelf. The continental shelves provide some of the richest fisheries known. At the end of the shelf, the slope drastically steepens (1:20), giving rise to the continental slope, and eventually the continental rise regions. This region averages approximately 4,000 m (13,120 ft) in vertical extent from the shelf to the abyssal plain. The ocean basin constitutes the most extensive area of the ocean bottom surface. Depths in this region range from 3,000 m to 6,000 m (9,840 to 19,680 ft). About 75 percent of the ocean floor is classified as basin area. The deepest areas of the ocean basins are the deep sea trenches, contrasted by the mid-oceanic ridges, which provide relative high points in the ocean bottom surface topography (Pickard and Emery, 1982).

Marine ports are generally located at the confluence of major rivers and the ocean. These regions are commonly referred to as estuaries, and provide a fragile habitat for much of the marine life found in the oceans. An estuary is defined as a semi-enclosed body of water with a free connection to the open ocean,

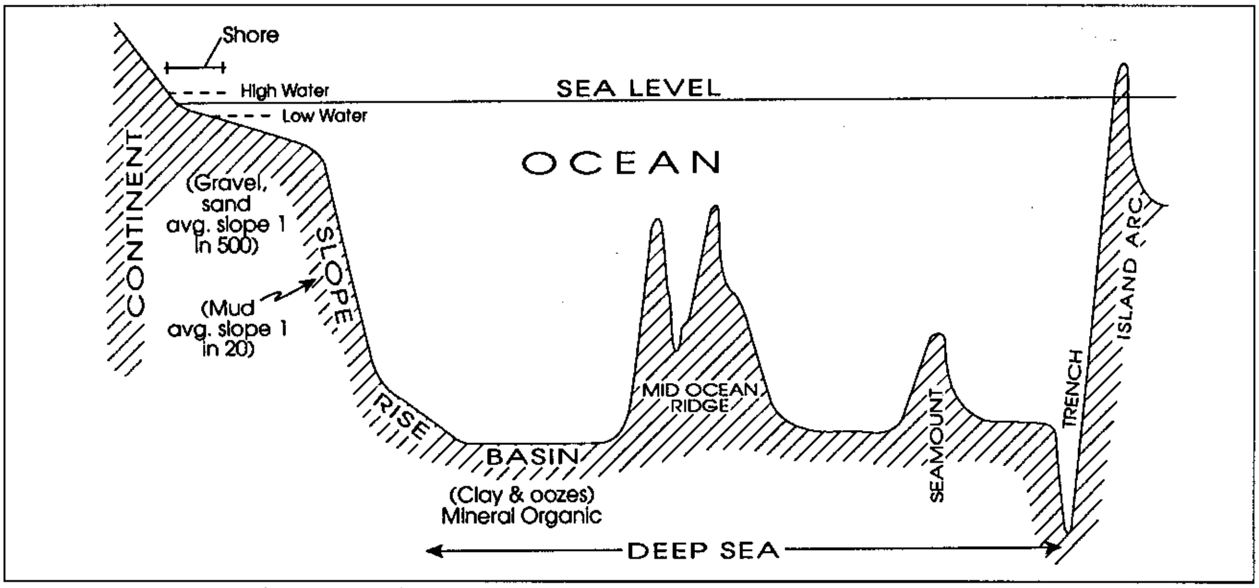


Figure 3-1 Schematic Section Across the Ocean Floor, Depicting Major Geological Features (Pickard and Emery, 1982)

where the saltwater is considerably diluted with freshwater. In general, the freshwater flowing into the estuary eventually exits the system in the upper (water) layer of the estuary, while the denser seawater enters the estuary through lower subsurface layers.

3.1.2 Chemical Oceanography

Seawater is a complex solution of minerals, salts, and elements, containing approximately 80 of the 92 naturally occurring elements. Hydrogen and oxygen, as water, constitute the largest elemental percent of seawater, with sodium chloride (NaCl) being the most abundant salt (78 percent) in the solution. Magnesium, calcium, and potassium chlorides and carbonates provide the bulk of the remaining constituents of the seawater solution. The ratio of these elements within the solution is relatively constant from ocean to ocean. However, in coastal areas where freshwater river influences are significant, the water chemistry can be substantially different. In addition to the major and minor constituents described above, trace metals, nutrient elements, dissolved atmospheric gases, and other organic matter also form important components of seawater. While trace metals are essential to the growth and development of certain organisms at low concentrations, these elements can become toxic when concentrated at high levels. Table 3-1 summarizes the concentration of major elements and trace elements, expressed in milligram per liter (mg/L), in seawater. The major nutrients (phosphates, silicates, and nitrates) provide the chief limiting agent for oceanic phytoplankton production. Atmospheric gases (e.g., oxygen and carbon dioxide) absorbed by the ocean play important roles in the overall global climate of the earth.

Naturally occurring radionuclides of uranium (such as ^{234}U, ^{235}U, ^{238}U), and polonium-210 (^{210}Po), are present in seawater, and in marine organisms, at concentrations generally greater than those found in terrestrial ecosystems. The ocean water concentrations of uranium isotopes are: ^{234}U, 1.30 picocuries per liter (pCi/l); ^{235}U, 0.05 pCi/l; and ^{238}U, 1.2 pCi/l (IAEA, 1976). For comparison, other major radioisotopes found in ocean water are: potassium-40 (40K), 486 pCi/l; thorium-232 (232Th), 540 pCi/; tritium (3H), 3 pCi/l; rubidium-87 (87Rb), 3 pCi/l; and Carbon-14 (14C), 1.8 pCi/l (IAEA, 1988).

Table 3-1 Concentration of Major Elements and Trace Elements in Seawater (CRC, 1991)

Element	mg/L	Trace Element	mg/L
Chlorine	19,000	Strontium	8.1
Sodium	10,500	Arsenic	0.003
Magnesium	1,350	Iron	0.01
Sulphur	885	Copper	0.003
Calcium	400	Zinc	0.01
Potassium	380	Cesium	0.0005
Bromine	65	Uranium	0.003
Fluorine	1.3	Lead	0.00003
Iodine	0.06	Zirconium	0.000022

The relationship between environmental concentrations of radionuclides and the concentration found in organisms is important in the study of food chain effects. Bioamplification, the increase in concentration of radionuclides in organisms progressively further up the food chain (as with organic pesticides in terrestrial environments), is observed in marine food chains. In the marine environment, uranium has not been found to bioamplify in fish, and there is only slight bioamplification in crustaceans and mollusks (IAEA, 1976). The readiness with which other radionuclide constituents of spent nuclear fuel may enter the food chain is variable, but generally low.

3.1.3 Physical Oceanography

The science of physical oceanography involves the development of a systematic quantitative description of ocean characteristics and circulations. Ocean circulations include not only the major, permanent ocean features (e.g., the Gulf Stream) that circulate continuously with fluctuating velocity and position dynamics, but also the smaller-scale circulation features (e.g., tides, waves, coastal currents, etc.). Gradients in temperature, salinity, and seawater density give rise to vertical and lateral circulations.

The primary forces behind the generation and maintenance of surface currents in the world ocean are the winds in the lower portions of the atmosphere. Low-level winds generate stresses on the ocean surface that give rise to the surface currents. However, these currents only affect the uppermost layers of the ocean. Thus, the global wind patterns establish the direction and magnitude of the surface currents. Figure 3-2 depicts the major components of the wind-induced surface circulation of the world ocean.

Northern hemisphere ocean basins are characterized by strong western basin boundary currents that transport warmer, less dense water poleward, and are balanced by weaker, colder return flows along the eastern basin boundaries. Examples of these flows in the northern hemisphere are the Gulf Stream and Kuroshio currents, and the Canary and Californian currents, respectively. These permanent circulation features are the result of the strong mid-latitude westerly winds and the easterlies in the tropics. Due to the strength of these oceanic and atmospheric circulations, North Atlantic and North Pacific shipping routes tend to follow these flows.

Also of interest are the deep water convective circulations, which are linked with the surface system circulation. In general, these circulations are generated in high latitudes by air-sea interaction processes producing relatively cold and dense surface waters that sink and flow into the central ocean basins. This loss of water in the high latitudes is replaced by warmer surface waters migrating poleward at intermediate depths. Thus, in considering the overall environmental impact of the proposed and alternative actions, the intermediate and bottom water masses/circulations cannot be ignored, due to their surface origin.

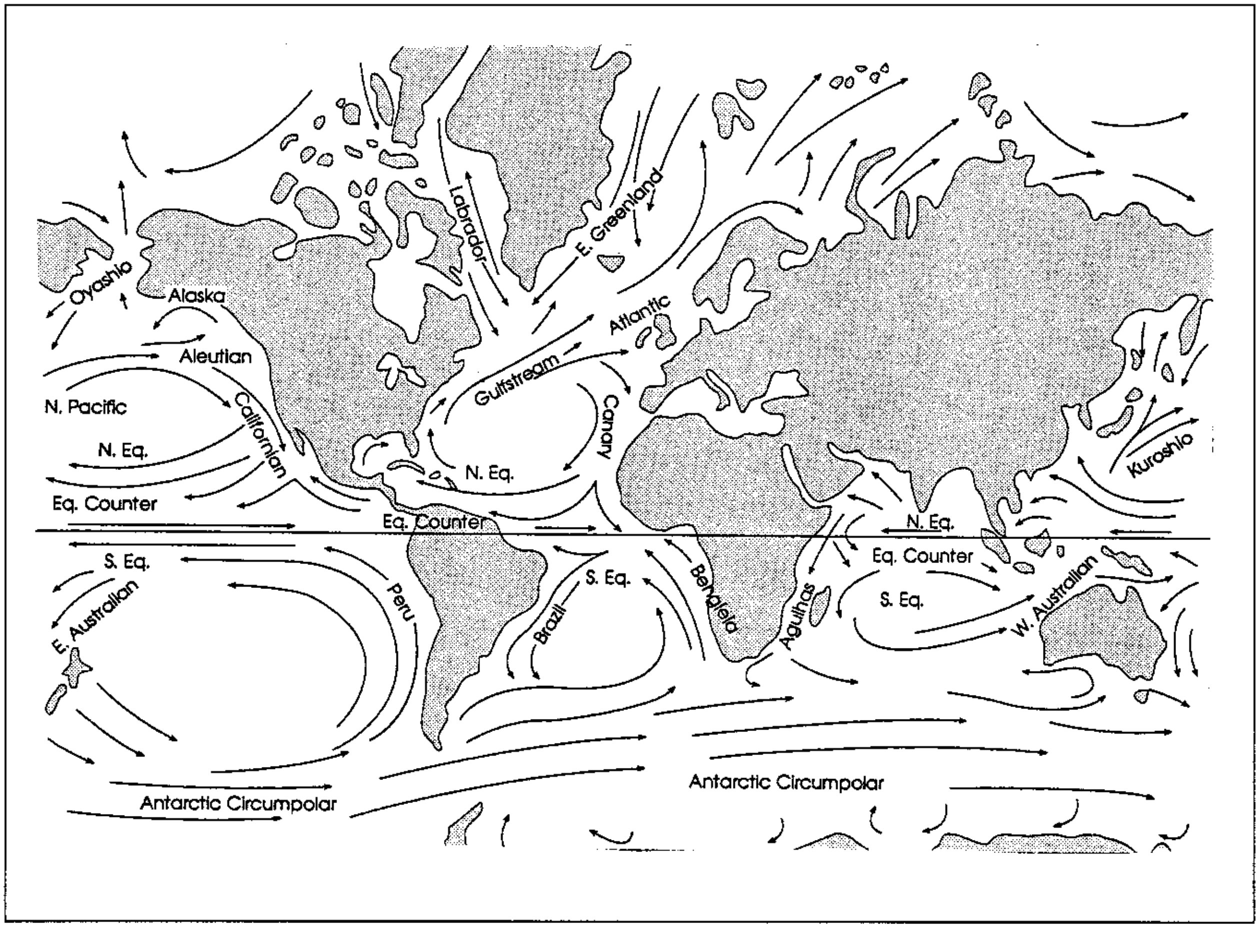


Figure 3-2 Major Wind-Driven Surface Currents of the World Ocean (Kennett, 1982)

3.1.4 Biological Oceanography

Biologically, the characteristics of ocean organisms dramatically change with ocean depth. Changes in organisms can be correlated with the decrease in the amount of light and the wavelength of the light that penetrate to a given depth. This variation in light is also influenced by the turbidity of the oceanic waters, and has a great influence on the biological productivity of a given region. Upper water layers are rich in nutrients and more productive than water layers found at depths greater than 200 m (660 ft). Abundant plant life supports the many animal species found at depths less than 200 m (660 ft). The estuarine areas found at the margins of the shelf region and the continents provide rich, productive breeding and spawning grounds for many marine organisms. In contrast, the deep ocean bottoms are limited in productivity because of the absence of light and the scarcity of nutrients (Friedrich, 1969).

The deep sea bottom dwellers are highly diverse, with many biological groups represented by more species than in most shallow-water communities (Hessler and Sanders, 1967). However, the number of individual organisms in a given volume does decrease in the deep sea and this, together with a general tendency toward decrease in the average size of the organisms, results in a dramatic reduction in standing stock or biomass on the deep ocean floor. In round figures, the total wet weight of bottom-living organisms in and on each square meter (m) of seabed decreases from 10-100 grams (g) on the continental shelf, to 1-10 g on the continental slope, and to only 0.1-1.0 g on the abyssal plain (Rice, 1978).

3.2 Individual Port Marine Environments

This section presents general environmental information for ten U.S. ports that have been identified as potential ports of entry. The ten ports are:

> Charleston, SC [includes the Naval Weapons Station (NWS) at Charleston and the Wando Terminal]; Galveston, TX; Hampton Roads (includes terminals at Newport News, Norfolk, and Portsmouth), VA; Jacksonville, FL; the Military Ocean Terminal at Sunny Point (MOTSU), NC; the NWS at Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA; and Wilmington, NC.

These ports are more fully described in Appendix D of this Environmental Impact Statement (EIS). Appendix D identifies the ports that were considered as potential ports of entry for foreign research reactor spent nuclear fuel, the criteria used in the port evaluation process, the method of evaluation, and the results of the evaluation process. Appendix D also presents population data for ports and transportation routes considered in the evaluation. Potential overland and barge transportation routes are described in Appendix E, and Appendix C presents information on the environmental impacts of marine transport.

The various policy, management, and implementation alternatives being considered in this EIS do not involve any construction or modification of port facilities, nor would the use of one or more ports for the receipt of foreign research reactor spent nuclear fuel be expected to noticeably increase the number of vessel calls to the port or interfere with existing port operations. Once at the port of destination, the spent nuclear fuel would be transferred from the vessel to a waiting truck or train and shipped to the destination as expeditiously as possible.

3.2.1 Environmental Information for the Potential Ports of Entry

This section presents summary environmental information for the potential ports of entry for foreign research reactor spent nuclear fuel.

3.2.1.1 Charleston, SC (Includes Terminals at the Naval Weapons Station and the Wando Terminal)

Charleston is the largest port city in South Carolina, and the greater Charleston area is one of the major seaports on the East Coast of the United States. The city of Charleston is located at the confluence of the Cooper and Ashley Rivers, approximately 11 km (7 mi) west of the Atlantic Ocean. The principal wharves are along the west bank of the Cooper River, except for the Wando Terminal, which is along the east bank of the Wando River near the city of Mount Pleasant, about 20 km (12 mi) from the Atlantic Ocean. The city of Charleston is on a peninsula, bounded on the west and south by the Ashley River and on the east by the Cooper River. In general, the elevation of the area ranges from sea level to approximately 6 m (20 ft) on the peninsula.

Environmental Conditions: The State of South Carolina has classified the water quality of the lower portion of the Wando River as both SFH and SA (SFH waters are shellfish harvesting waters, and SA waters are suitable for primary and secondary recreation and for other water uses requiring lower water quality). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for James Island, SC, the Wando Terminal and the NWS Charleston are located in a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Charleston harbor which is traversed enroute to either terminal, is located in a high-salinity estuarine habitat (generally 16.5 to 30 ppt) (FWS, 1980a).

The State of South Carolina has classified the water quality of the portion of the Cooper River above the confluence with the Ashley River as SB (SB waters are tidal saltwaters suitable for secondary contact recreation, crabbing, and fishing, except the harvesting of clams, mussels, or oysters for market purposes and human consumption). The waters of Goose Creek, upstream of the confluence with the Cooper River to the dam at the Charleston Waterworks, are also Class SB (Department of the Navy, 1994).

State or Federally protected endangered or threatened aquatic species in the vicinity of the Charleston harbor include the shortnose sturgeon, Atlantic sturgeon, and the American shad. Bachman's warbler is a Federally protected bird species also found in the vicinity (FWS, 1980a). While there are some wetlands in the vicinity of Wando Terminal and on the property of NWS Charleston (Department of the Navy, 1990 and 1994), there are no known special wildlife sanctuaries or habitats of concern in the general area. Bald eagles have been observed on the NWS Charleston property and are believed to be nesting in the far northern areas of the Station. Red-cockaded woodpeckers are known to inhabit NWS Charleston. Although, the hurricane Hugo (September 1989) destroyed much of their habitat (mature pine trees with red heart disease), several colonies are surviving with the assistance of artificial nest bates (Lewis, 1995). The Charleston harbor area and the west bank of the Cooper River are commercially well developed.

The lower Wando and Cooper Rivers and the Charleston harbor support a large number of aquatic and terrestrial species. Aquatic species commonly found in the vicinity include crabs, oysters, clams, shrimp, sturgeon, herring, shad, seabass, kingfish, drum, flounder, and mackerel. Marine mammals, including dolphins and whales, have been sighted in the harbor. According to the South Carolina Heritage Trust, no rare, threatened, or endangered species or communities have been recorded in the area near the Wando Terminal (McBee, 1994).

Climatic Conditions: The climate of this region is temperate, primarily due to its close proximity to the Atlantic Ocean. The prevailing winds are generally northerly in the fall and winter months, becoming more southerly during the summer months. This type of circulation serves to "warm" the region during winter and "cool" it during the summer. Summer is the rainy season in Charleston, with the city receiving 41 percent of the annual total rainfall during the summer months. Except for the occasional tropical storm or hurricane, the majority of this rain occurs during afternoon and evening thunderstorms. The late summer and early fall brings the highest probability of tropical storm activity to the Charleston area. The fall season is a transitional period, where temperature extremes are rare and sunshine is abundant. The winters in this area are mild with periods of rain. However, in contrast to the summer, winter rains tend to be steady and uniform, and last for several days. The most unstable period in this region is spring, when the confluence of warm moist tropical air and cool dry continental air increase the occurrence of severe weather in this region. The average earliest freeze is in early December, and the average last frost is in late February (NOAA, 1992c).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Charleston, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 m per hour). The greater Charleston area is located in a moderate seismic zone with an acceleration of 0.15 g. The effective peak velocity-related acceleration represents the back-and-forth horizontal motion of the ground due to a seismic event at a period of 1.0 sec. This acceleration is expressed in relation to g, where g equals acceleration due to gravity.

Naval Weapons Station - Charleston: The NWS is located on the west bank of the Cooper River, north of the city of North Charleston. The NWS is approximately 7080 hectares (17,500 acres) in size and is located in southeastern Berkely County, South Carolina, about 30 km (19 mi) from the Atlantic Ocean. The NWS has two useful wharves and two useful piers. Wharf Alpha and Pier Bravo have cranes and are capable of loading trucks or trains directly from the ships. Pier Charlie and the Military Traffic Management Command Terminal would have to use shipboard or mobile cranes to load trucks. Several facilities on the NWS could be used to transfer spent fuel casks or containers from trucks to rail cars. A map of the port is shown in Figure 3-3.

The 1990 population within 16 km (10 mi) of the Wharf Alpha was 209,188. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five U.S. Department of Energy (DOE) management sites are: the Savannah River Site, 46,200; the Oak Ridge Reservation, 108,000; the Idaho National Engineering Laboratory, 498,000; the Hanford Site, 550,000; and the Nevada Test Site, 540,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 303 km (188 mi), the Oak Ridge Reservation, 647 km (402 mi), the Idaho National Engineering Laboratory, 3,930 km (2,442 mi), the Hanford Site, 4,601 km (2,859 mi), and the Nevada Test Site, 4,094 km (2,544 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-4 shows the ethnic composition for the area surrounding the port at the NWS Charleston. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted about 31 percent of the total population, and approximately 88 percent of the minority population for the area surrounding the port. Figure 3-5 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

Wando Terminal: This South Carolina State Port Authority terminal is located at the confluence of the Wando and Cooper rivers, on the east bank of the Wando River, near the incorporated city of Mount Pleasant. The facility has three modern container berths, with a fourth under construction, and a large paved container storage yard. The Wando terminal is about 8.1 km (5 mi) from the nearest Interstate highway and 15 km (9 mi) from the nearest intermodal rail yard. A map of the port is shown in Figure 3-6.

The 1990 population within 16 km (10 mi) of the Wando Terminal was 233,424. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five U.S. Department of Energy (DOE) management sites are: the Savannah River Site, 65,700; the Oak Ridge Reservation, 127,000; the Idaho National Engineering Laboratory, 518,000; the Hanford Site, 569,000; and the Nevada Test Site, 559,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 327 km (203 mi), the Oak Ridge Reservation, 671 km (417 mi), the Idaho National Engineering Laboratory, 3,954 km (2,457 mi), the Hanford Site, 4,625 km (2,879 mi), and the Nevada Test Site, 4,118 km (2,559 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-7 shows the ethnic composition for the area surrounding the Wando Terminal. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-8 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

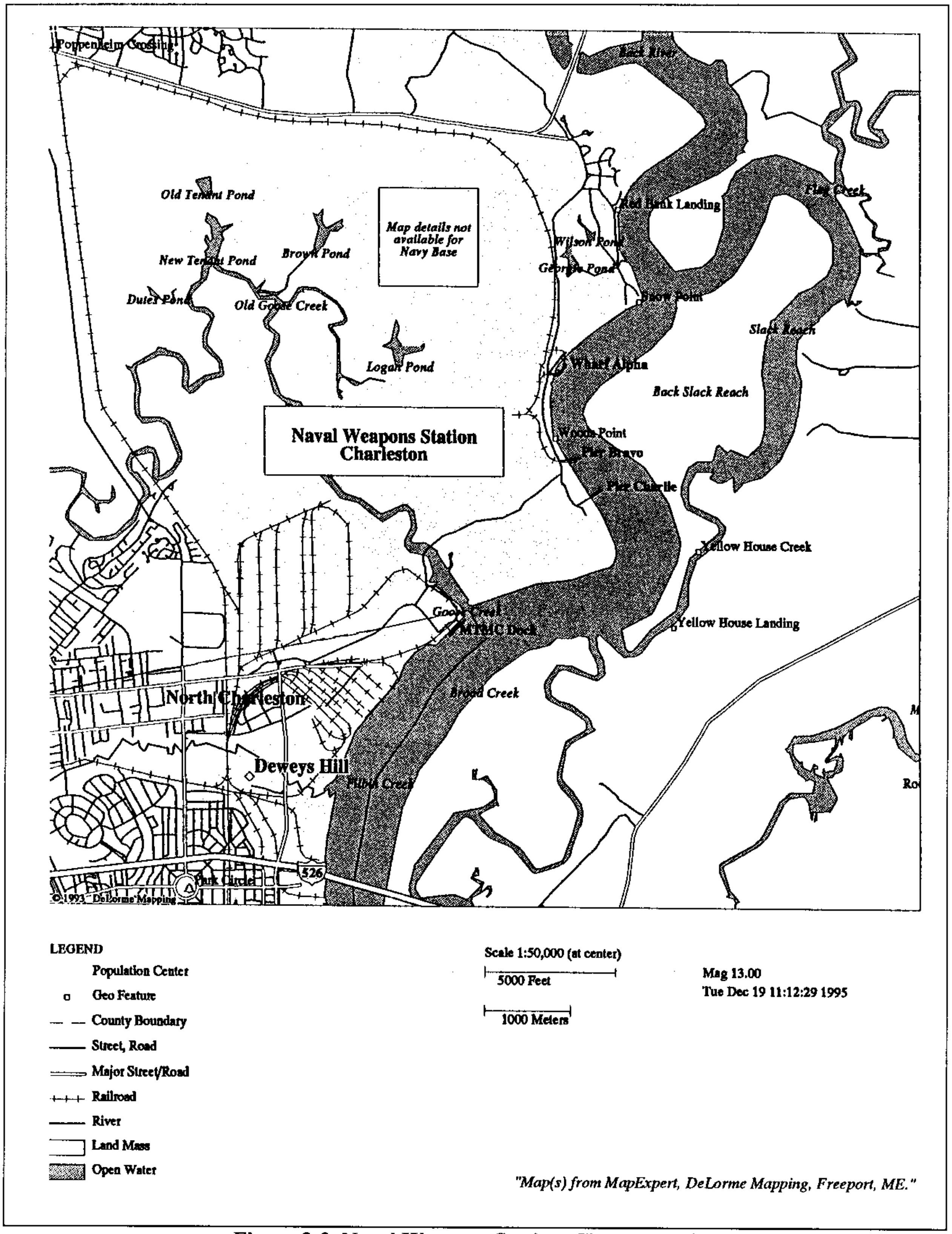


Figure 3-3 Naval Weapons Station, Charleston, SC

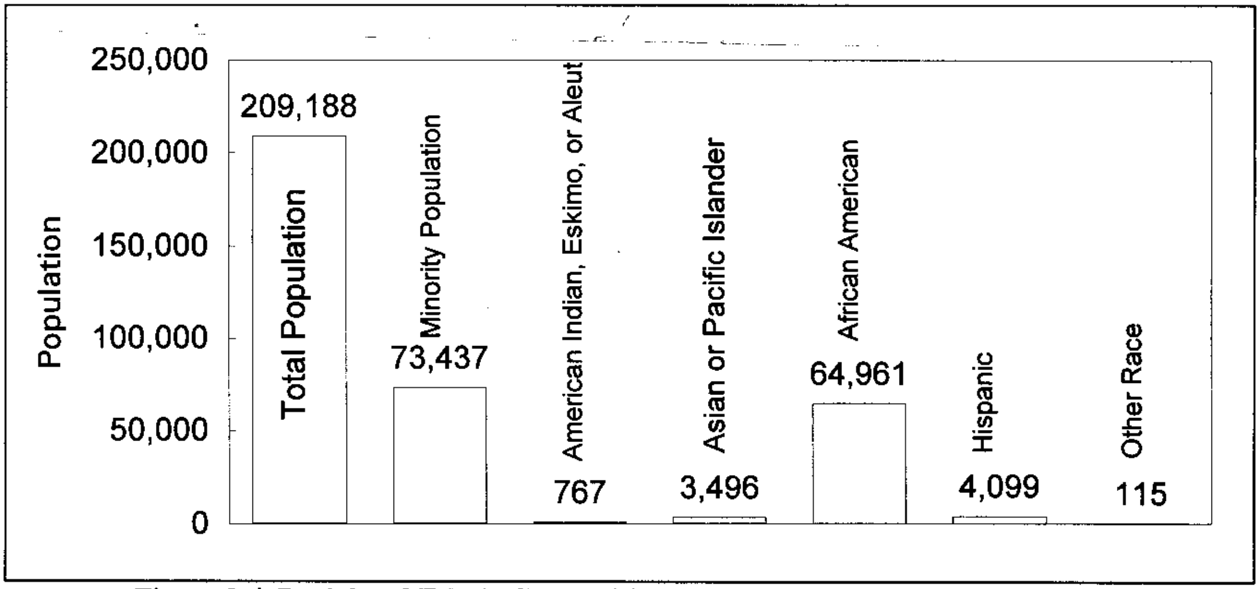


Figure 3-4 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Naval Weapons Station, Charleston

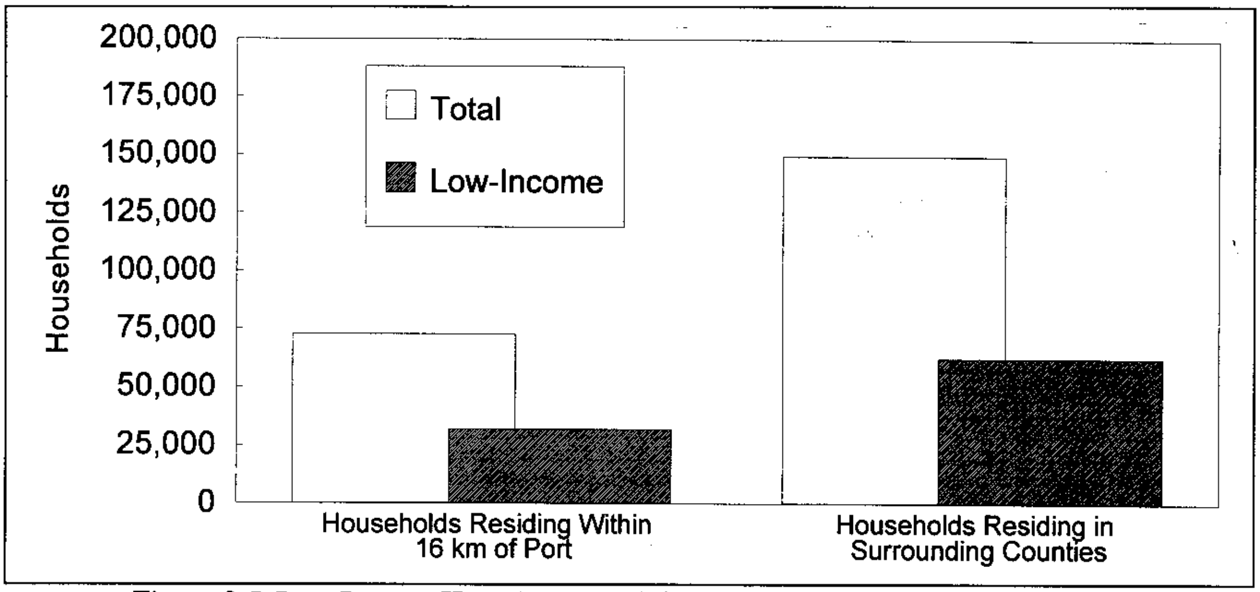


Figure 3-5 Low-Income Households Residing within 16 km (10 mi) of the Naval Weapons Station, Charleston

3.2.1.2 Galveston, TX

Galveston, TX is situated within 16 km (10 mi) of the entrance to the Gulf of Mexico. The city of Galveston occupies the entire width of the east end of Galveston Island. The shipping wharves are on the north side of the island and the Gulf of Mexico is on the south. The Port of Galveston is located in the heart of the city. A map of the port is shown in Figure 3-9.

Galveston is a major resort and tourist center for the Southwest. There is a waterfront tourist attraction at "Pier 21" close to the port area. A public park on Pelican Island, reached by causeway, is located across the Intracoastal Waterway from the Port of Galveston. A cruise ship terminal is located at Pier 25 in the

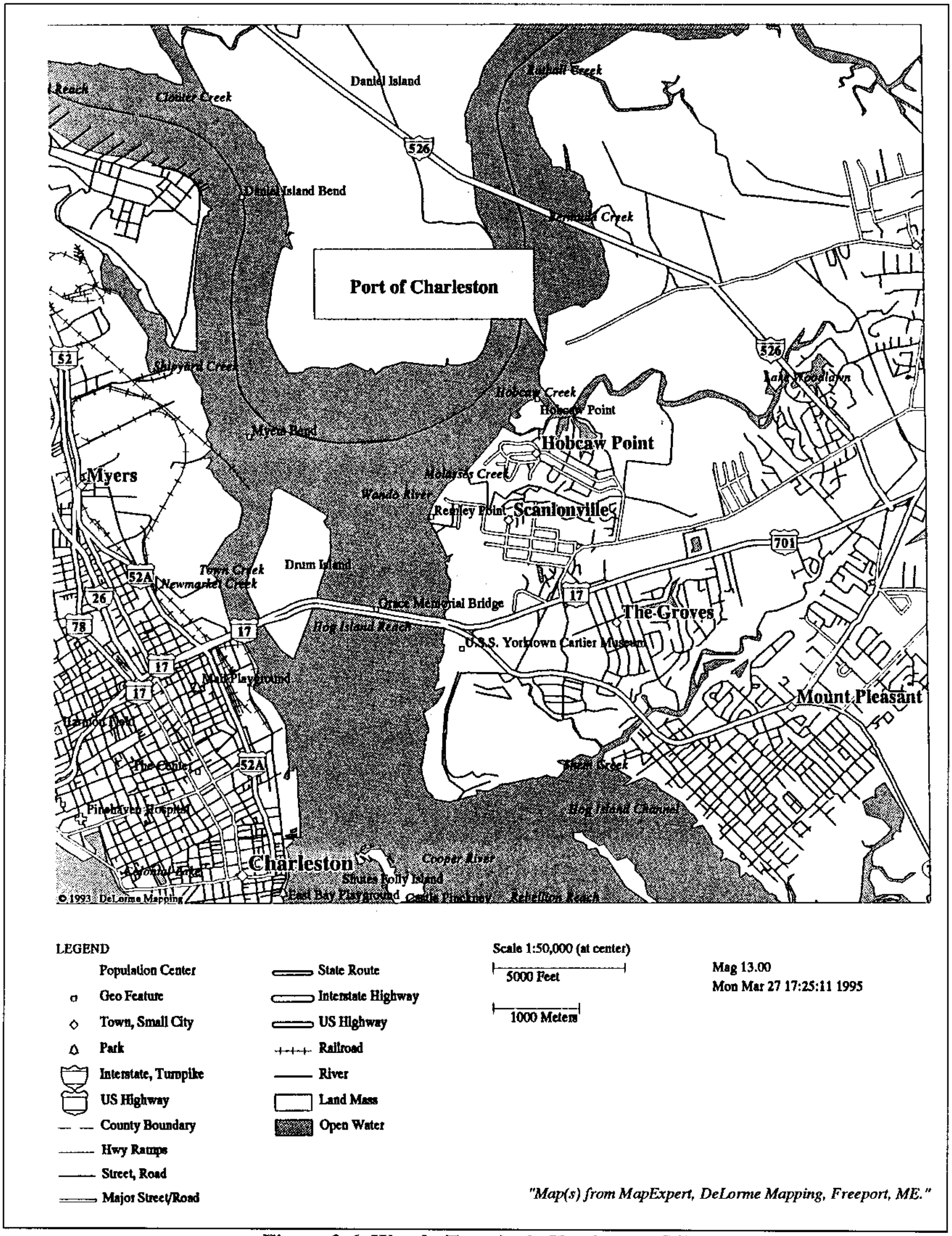


Figure 3-6 Wando Terminal, Charleston, SC

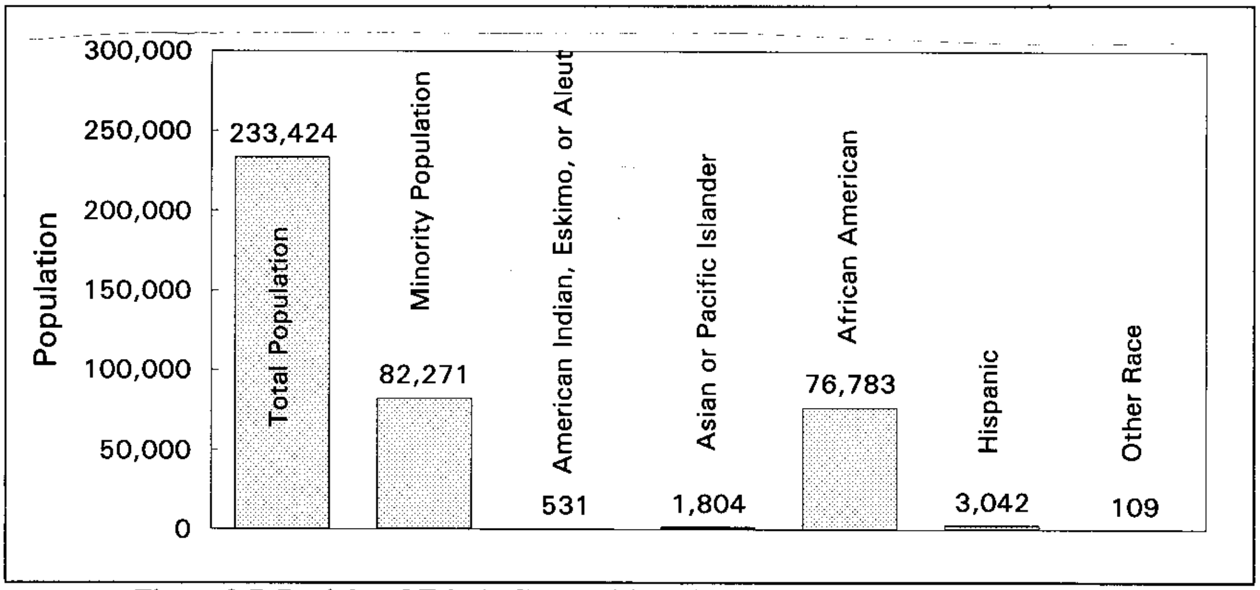


Figure 3-7 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Wando Terminal, Charleston

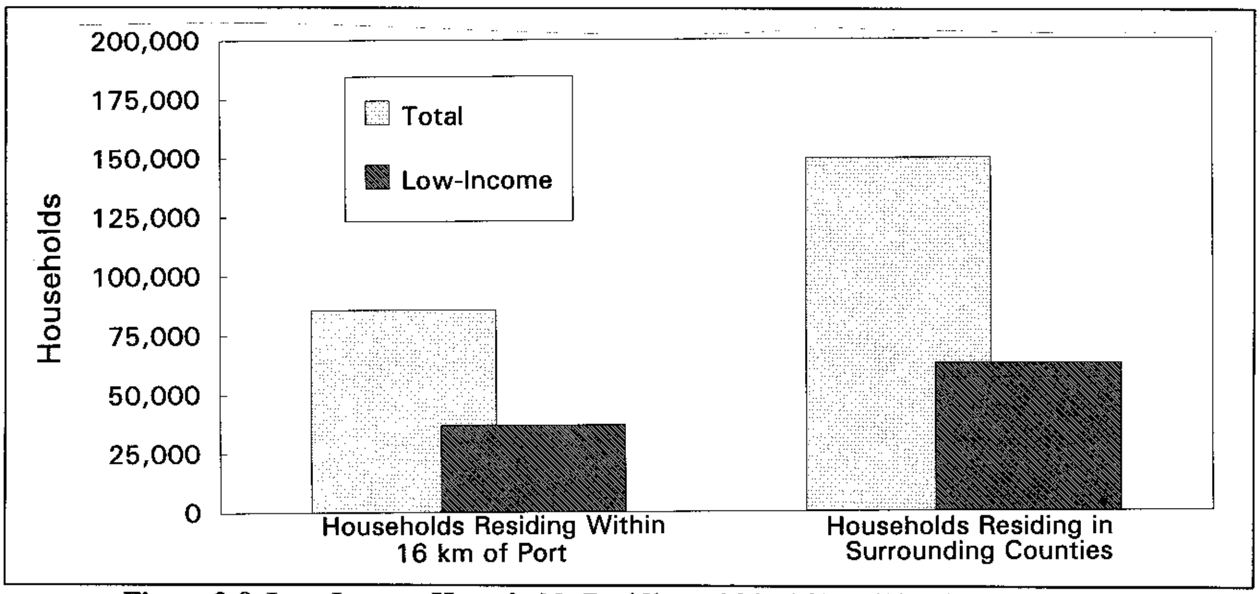


Figure 3-8 Low-Income Households Residing within 16 km (10 mi) of the Wando Terminal, Charleston

heart of the Port of Galveston complex, and there is a tanker terminal on Pelican Island across from the Port of Galveston at its southern end. A Federal project provides for an entrance channel, and an outer bar channel both dredged to 12.8 m (42 ft).

The Port of Galveston's principal container handling facility is the container terminal at Pier 10. This facility has a controlled all-weather truck entrance and interchange area. The terminal is connected to Interstate Highway 45 on the mainland by a 9.3 km (5.8 mi) four-lane State highway and two 2.8 km (1.75 mi) causeways that cross the southwest end of Galveston Bay. The container handling facility is served by four major railroads, the Burlington Northern, Santa Fe, Southern Pacific, and Union Pacific

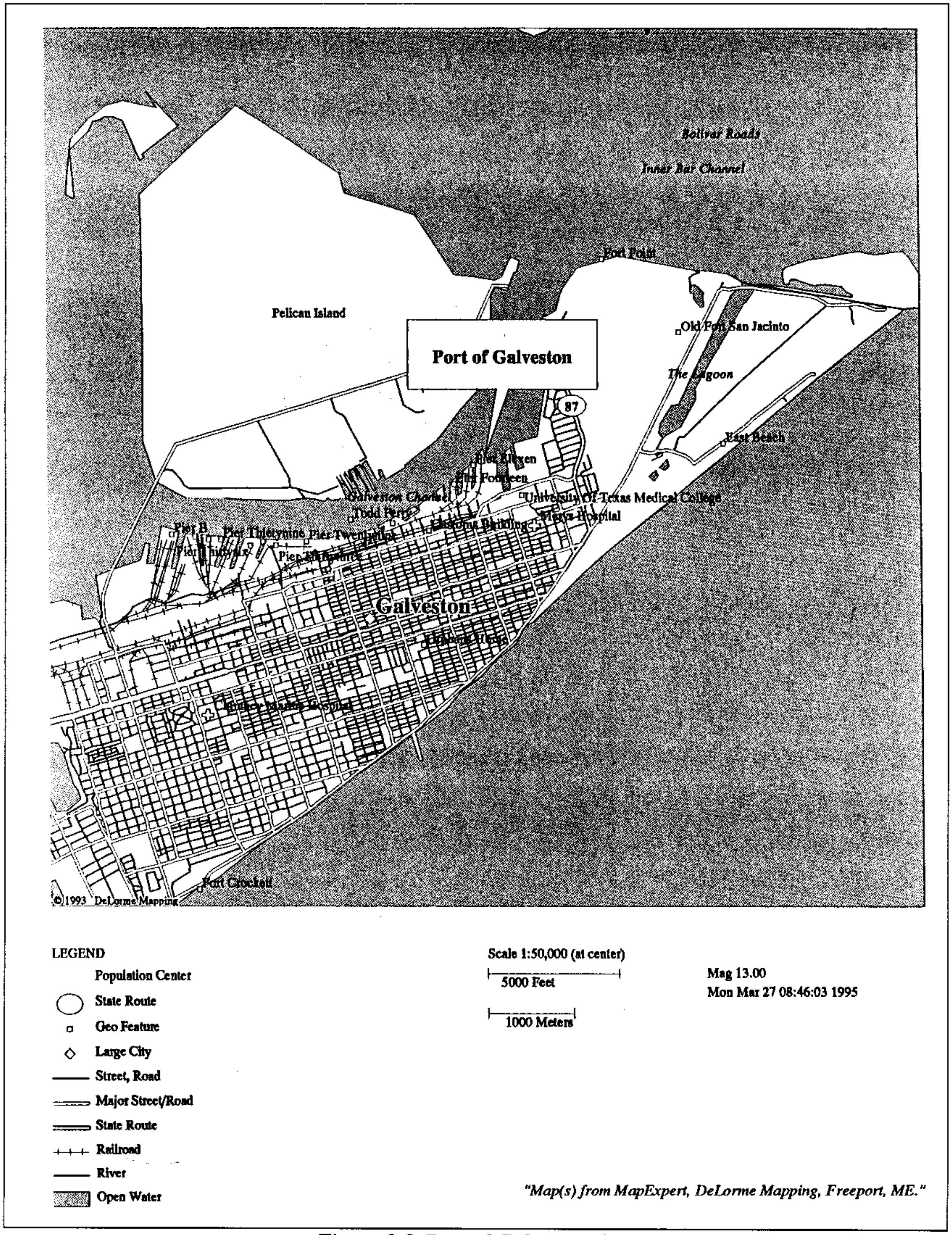


Figure 3-9 Port of Galveston, TX

Lines. Galveston Railway, Inc., provides terminal connections and performs switching of all port rail traffic. An intermodal container transfer terminal is located within the container terminal, and trackage extends to within 30.5 m (100 ft) of ship berths.

The 1990 census population within 16 km (10 mi) of the port terminals was 73,322. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 403,000; the Oak Ridge Reservation, 337,000; the Idaho National Engineering Laboratory, 526,000; the Hanford Site, 575,000; and the Nevada Test Site, 595,000. Populations along rail routes to these sites are slightly larger for the Savannah River Site and the Oak Ridge Reservation, but slightly less for the Idaho National Engineering Laboratory, the Hanford Site, and the Nevada Test Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 1,600 km (1,000 mi); the Oak Ridge Reservation, 1,550 km (963 mi), the Idaho National Engineering Laboratory, 3,070 km (1,908 mi); the Hanford Site, 3,740 km (2,324 mi); and the Nevada Test Site, 3,000 km (1,864 mi). Distances along rail routes are slightly longer.

Environmental Conditions: A large number of aquatic and terrestrial species frequent the Galveston Bay area. A variety of birds migrate, winter, and breed along the Texas Coast including shorebirds, songbirds, waterfowl and raptors (Breslin, 1993; FWS, 1992). These endangered/threatened bird species include the brown pelican, peregrine falcon, bald eagle, Atwater's greater prairie-chicken, piping plover, and the Eskimo curlew (State-threatened only). Endangered/threatened marine mammals and sea turtles also are found in the coastal bay systems and the Gulf of Mexico. Galveston Bay is within the range of the green, hawksbill, Kemp's ridley, leatherback, and loggerhead sea turtles. While no protected species are known to be located within the Port of Galveston, significant populations of the endangered brown pelican and the threatened piping plover exist nearby (Werner, 1994). The U.S. Fish and Wildlife Service reported that as many as 600 brown pelicans have been sighted loafing on the north end of Little Pelican Island, which is approximately 5.6 km (3.5 mi) northwest of the port. In addition, approximately 125 pairs nested and produced 90 young ones at this site in 1994. This was the first time that brown pelicans had successfully nested in Galveston Bay in over 40 years. Wintering populations of the threatened piping plover use the northeastern end of Galveston Island and the southeastern end of Bolivar Peninsula. Of the 3,187 birds observed during the first Gulf Coast count of wintering piping plovers, 1,904 were observed on the Texas coastline (Werner, 1994).

A great amount of commercial and recreational fishing occurs in Galveston Bay and the Gulf of Mexico. Shellfish are the most important commercial species, particularly shrimp followed by eastern oysters and blue crabs (TPWD, 1989a). The most valuable finfish landed from the Galveston Bay system are black drum and mullet. In 1988, a total of 507,7169 kg (11,169,773 lb) of shellfish valued at $13,489,146 was landed from the Galveston Bay System; a total of 224,536 kg (493,980 lb) of finfish valued at $226,140 was also landed. The major recreational species of fish that were caught in the Galveston Bay system in 1987-1988 were: Atlantic croaker, sand seatrout, spotted seatrout, southern flounder, black drum, and red drum (TPWD, 1989b). Galveston Bay has been named as an "estuary of national significance" by the U.S. Congress. The implementation of the proposed action would pose no significant radiological or non-radiological risks to the environment in the Galveston area, including estuaries.

Climatic Conditions: The climate of the Galveston area is predominantly marine, with periods of modified continental influence during winter. The port is subject to hurricanes and tropical storms (NOAA, 1993a). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For the Port of Galveston, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Ethnic and Income Characteristics: Figure 3-10 shows the ethnic composition for the area surrounding the Port of Galveston. This figure shows the population residing within 16 km (10 mi) of the port, according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 27 percent of the total population, and approximately 54 percent of the minority population for the area surrounding the port. Hispanics made up about 20 percent of the total population, and approximately 40 percent of the minority population around the port. Figure 3-11 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

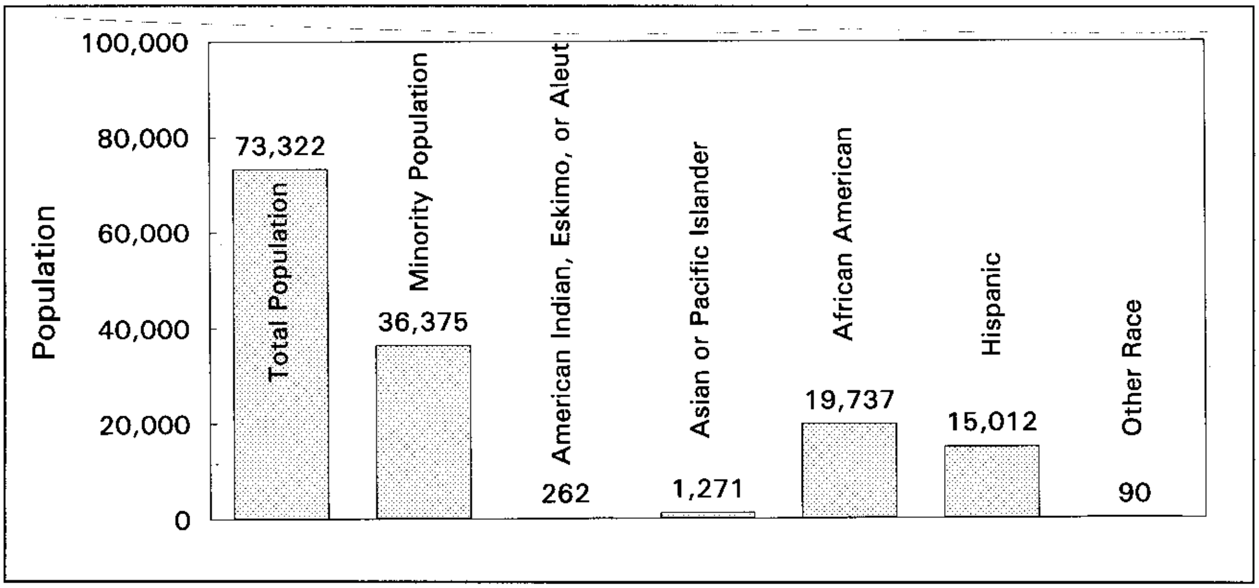


Figure 3-10 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Galveston

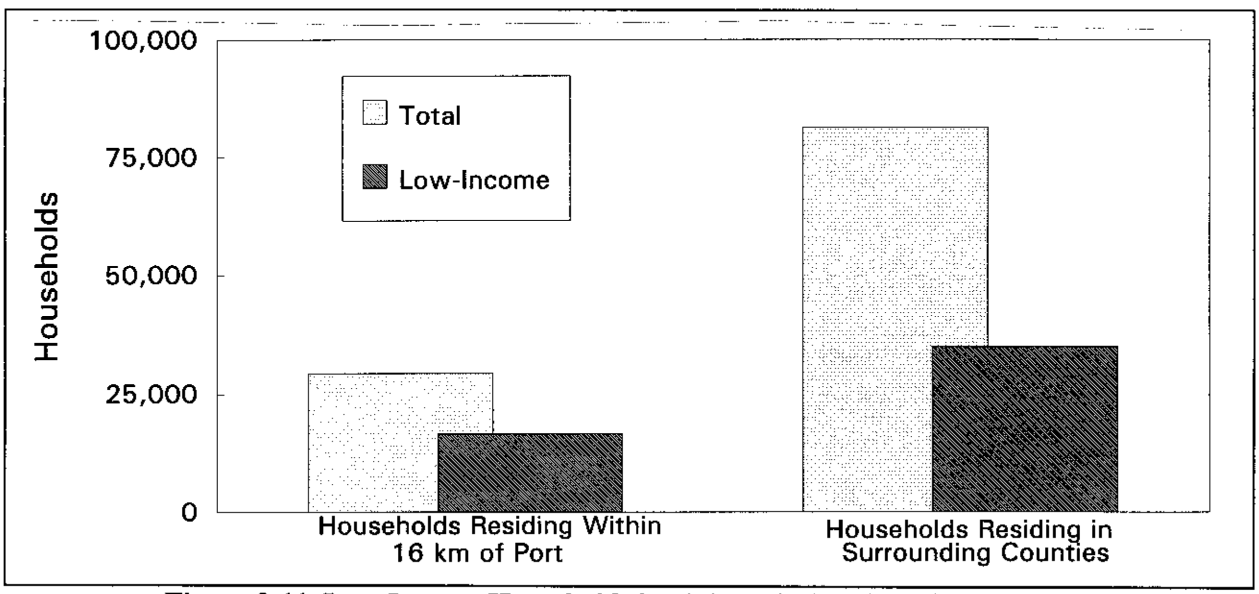


Figure 3-11 Low-Income Households Residing within 16 km (10 mi) of the Port of Galveston

3.2.1.3 Hampton Roads, VA (Includes Terminals at Newport News, VA; Norfolk, VA; and Portsmouth, VA)

Hampton Roads is one of the world's foremost bulk cargo harbors, and has more collective experience handling spent nuclear fuel than any other port in the United States. It is a multi-terminal port with privately and publicly owned marine cargo handling facilities, and is located at the southwest corner of the Chesapeake Bay at the confluence of the James and the Elizabeth Rivers. The port is about 26 km (16 mi) from the Virginia Capes and the entrance from the Atlantic Ocean. The major terminals located on the Elizabeth and James Rivers are approximately another 10 to 13 km (6 to 8 mi) from the Chesapeake Bay. The port includes the port terminals at Norfolk, Portsmouth, and Newport News. All three terminals are located in commercial port districts of their respective cities, somewhat separated from other community activities, in areas dedicated primarily to port industrial usage. Adjacent communities include the cities of Chesapeake and Virginia Beach.

Environmental Conditions: The Port of Hampton Roads is located at the confluence of the James River and the Chesapeake Bay, approximately 29 km (18 mi) west of the Atlantic Ocean. The average elevation of this region is approximately 4 m (13 ft) above sea level. There are no known areas of special environmental concern other than the growing interest in preservation of the Chesapeake Bay and its tributary rivers. The Dismal Swamp National Wildlife Refuge is located about 16 km (10 mi) from the two terminals on the Elizabeth River, but water drainage from the swamps is toward the port area. The swamp refuge is far enough from the terminals that potential negative impacts of low-probability, severe accidents in the ports on wildlife populations would be negligible. The three port terminals at Hampton Roads are described separately below.

Climatic Conditions: The geographic location of this region is especially favorable, tending to be located south of the predominant winter extratropical cyclone tracks which originate at higher latitudes and north of the usual tropical cyclone (e.g., tropical storms and hurricanes) paths. In general, the winters are mild with slightly warmer temperatures during the spring and fall seasons. The summer season is warm and long, but is characterized by frequent cool periods, generated by cool northeasterly winds off of the North Atlantic. Extreme cold waves are infrequent, and temperatures below -18°C (0°F) are almost nonexistent. In general, winters pass without measurable snowfall and most snowfall melts within 24 hours. The average first sub-freezing day in the fall is November 17th and the last occurrence in the spring is March 23rd. The predominant wind directions since 1984 are from the south-southwest and north-northeast and vary seasonally (NOAA, 1992c).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Hampton Roads, the Uniform Building Code provides a basic wind speed of about 140 km per hour (90 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Newport News Marine Terminal: This terminal is located on the north shore of the Port of Hampton Roads on the James River. It is a combination container, roll-on/roll-off, and breakbulk terminal. The facility has two piers, two container vessel berths, and four container cranes. There is covered storage on both piers. A map of the Newport News Marine Terminal is shown in Figure 3-12.

The 1990 population within 16 km (10 mi) of the port terminal was 430,757. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 181,000; the Oak Ridge Reservation, 209,000; the Idaho National Engineering Laboratory, 628,000; the Hanford Site, 677,000; and the Nevada Test Site, 691,000. Populations along rail

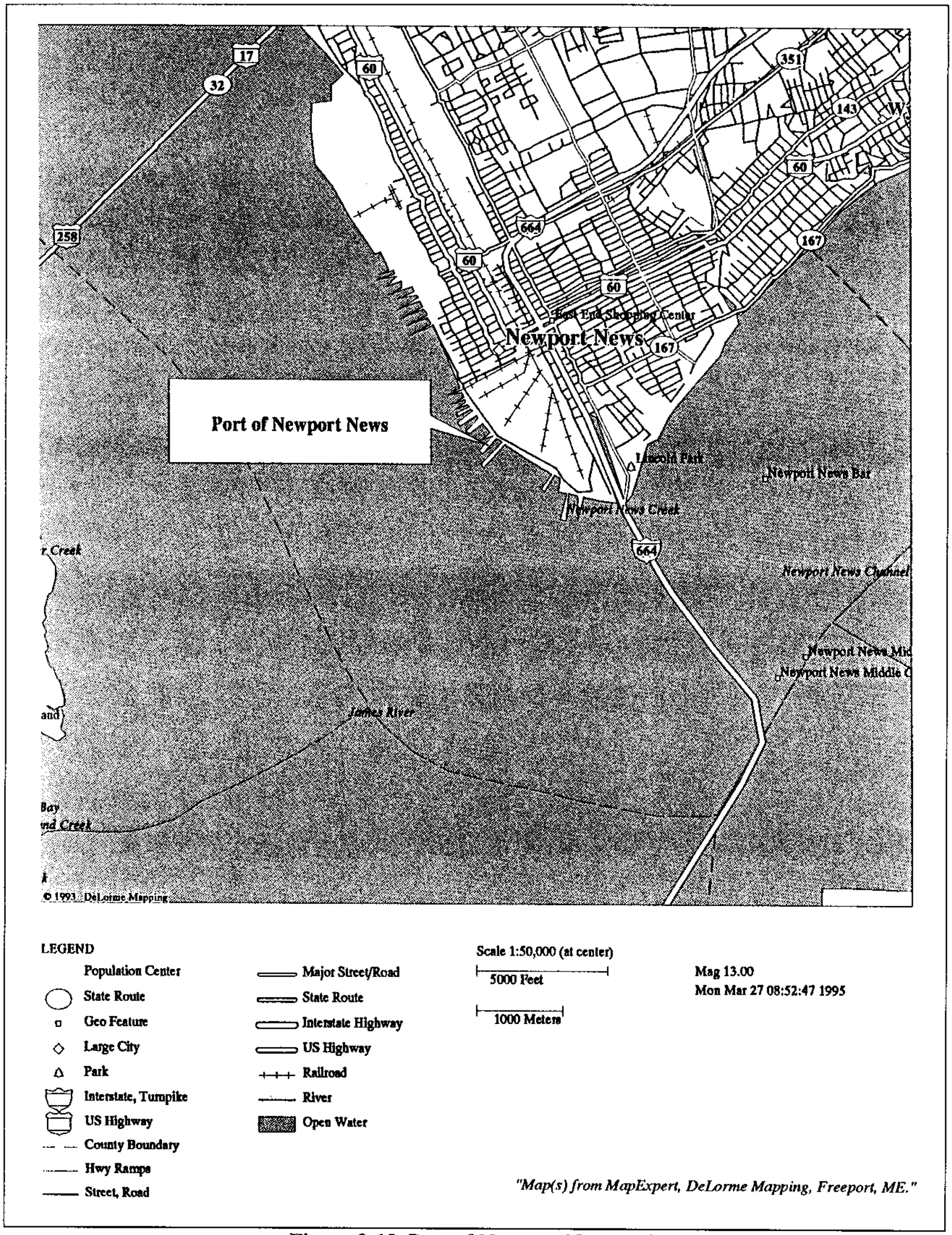


Figure 3-12 Port of Newport News, VA

routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 840 km (522 mi); the Oak Ridge Reservation, 890 km (553 mi); the Idaho National Engineering Laboratory, 4,010 km (2,492 mi); the Hanford Site, 4,680 km (2,908 mi); and the Nevada Test Site, 4,172 km (2,592 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-13 shows the ethnic composition for the area surrounding the port at Newport News. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 32 percent of the total population, and approximately 86 percent of the minority population for the area surrounding the port. Figure 3-14 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

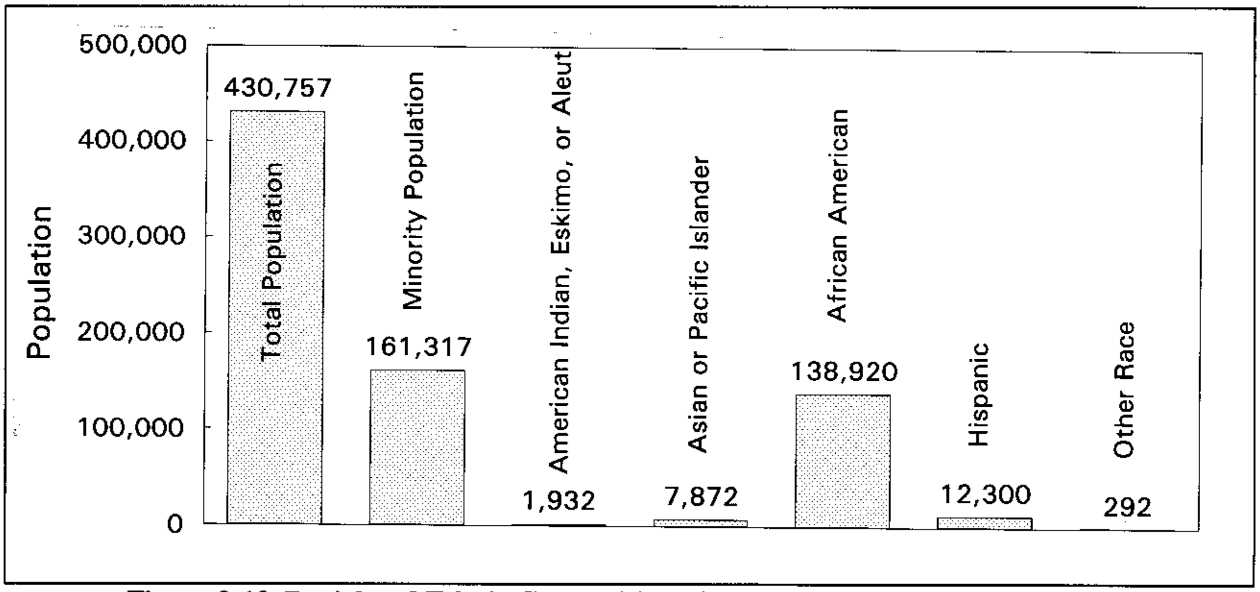


Figure 3-13 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Newport News

Norfolk International Terminal: This terminal is located on the south side of the Port in Norfolk, adjacent to the Navy Base on the Elizabeth River Channel. Norfolk International Terminal has 4 container vessel berths, 7 container cranes, a roll-on/roll-off berth, and covered pier storage. Sewell's Point Terminal, located at the north end (seaward) of Norfolk International Terminal's container berths has two piers, and covered storage for breakbulk cargoes. A map of Norfolk International Terminal is shown in Figure 3-15.

The 1990 population within 16 km (10 mi) of the port terminals was 681,864. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 131,000; the Oak Ridge Reservation, 174,000; the Idaho National Engineering Laboratory, 631,000; the Hanford Site, 694,000; and the Nevada Test Site, 694,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 800 km (497 mi); the Oak Ridge Reservation, 880 km (547 mi); the Idaho National Engineering Laboratory, 4,070 km (2,529 mi); the Hanford Site, 4,740 km (2,945 mi); and the Nevada Test Site, 4,240 km (2,635 mi). Distances along rail routes are slightly longer.

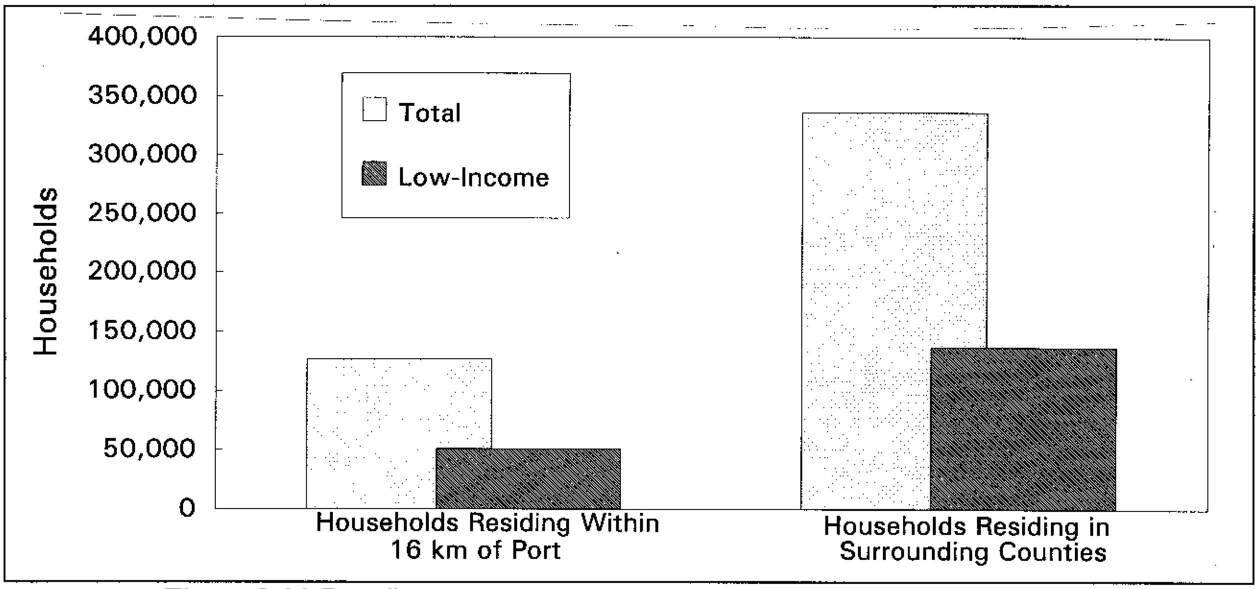


Figure 3-14 Low-Income Households Residing within 16 km (10 mi) of the Port of Newport News

Ethnic and Income Characteristics: Figure 3-16 shows the ethnic composition for the area surrounding the port at Norfolk, VA. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-17 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

Portsmouth Marine Terminals: Portsmouth Marine Terminals are located at the confluence of the Elizabeth River and its western branch in the city of Portsmouth. The terminals have 3 berths that handle container, breakbulk and roll-on/roll-off cargoes. The terminals have 3 container cranes, and more than 14,000 m^2 (150,000 ft^2) of warehouse space. A map of the Portsmouth Marine Terminals is shown in Figure 3-18.

The 1990 population within 16 km (10 mi) of the Portsmouth Marine Terminals was 665,700. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 135,000; the Oak Ridge Reservation, 257,000; the Idaho National Engineering Laboratory, 670,000; the Hanford Site, 718,000; and the Nevada Test Site, 732,000. Populations along rail routes to these sites are about the same for eastern sites and slightly larger for western sites. The distances to the five potential sites on interstate routes are: the Savannah River Site, 810 km (503 mi); the Oak Ridge Reservation, 780 km (485 mi); the Idaho National Engineering Laboratory, 4,040 km (2,510 mi); the Hanford Site, 4,710 km (2,927 mi); and the Nevada Test Site, 4,210 km (2,616 mi). Distances along rail routes are slightly longer.

Ethnic and Income Characteristics: Figure 3-19 shows the ethnic composition for the area surrounding the port at Portsmouth. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 89 percent of the minority population for the area surrounding the port. Figure 3-20 shows analogous information for

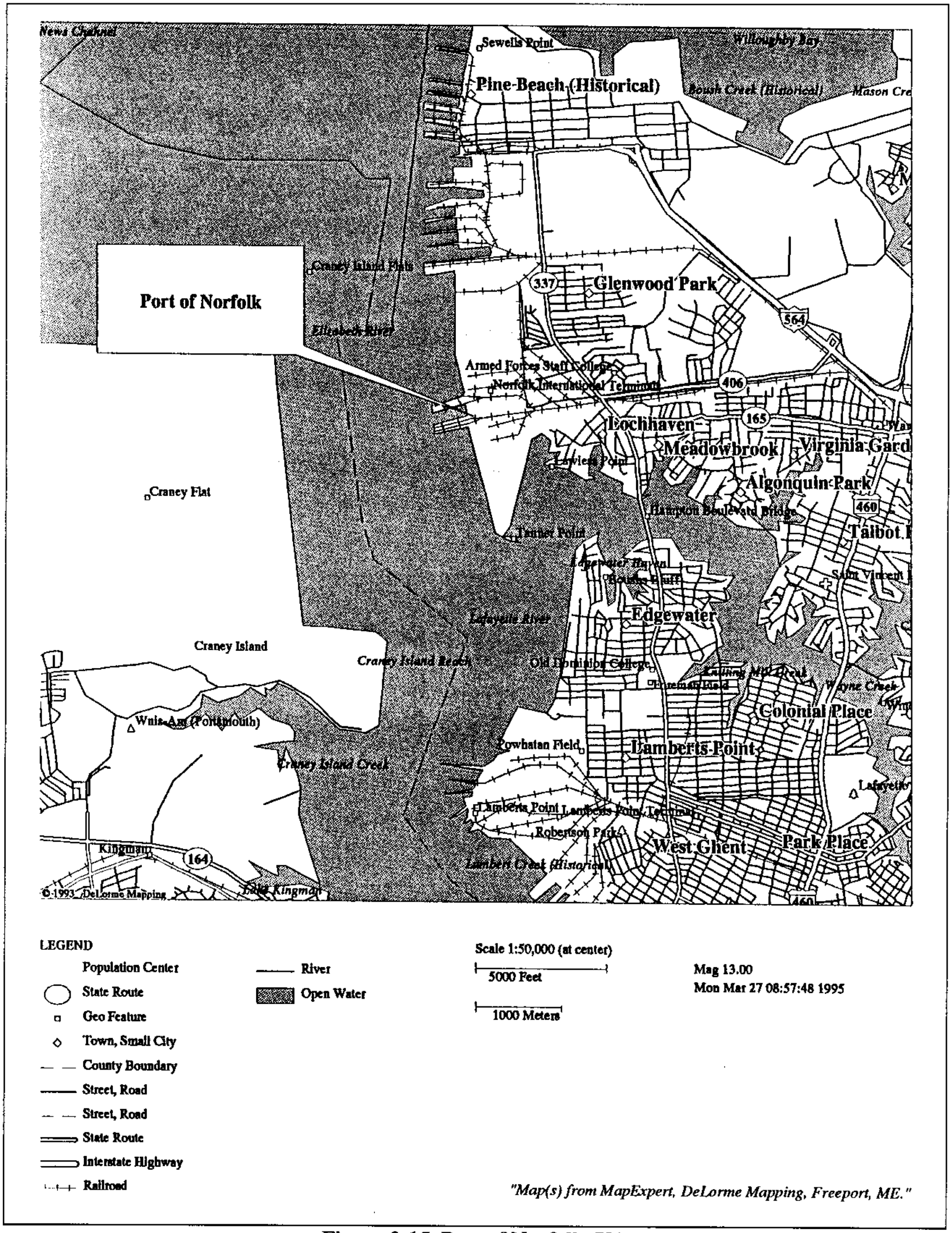


Figure 3-15 Port of Norfolk, VA

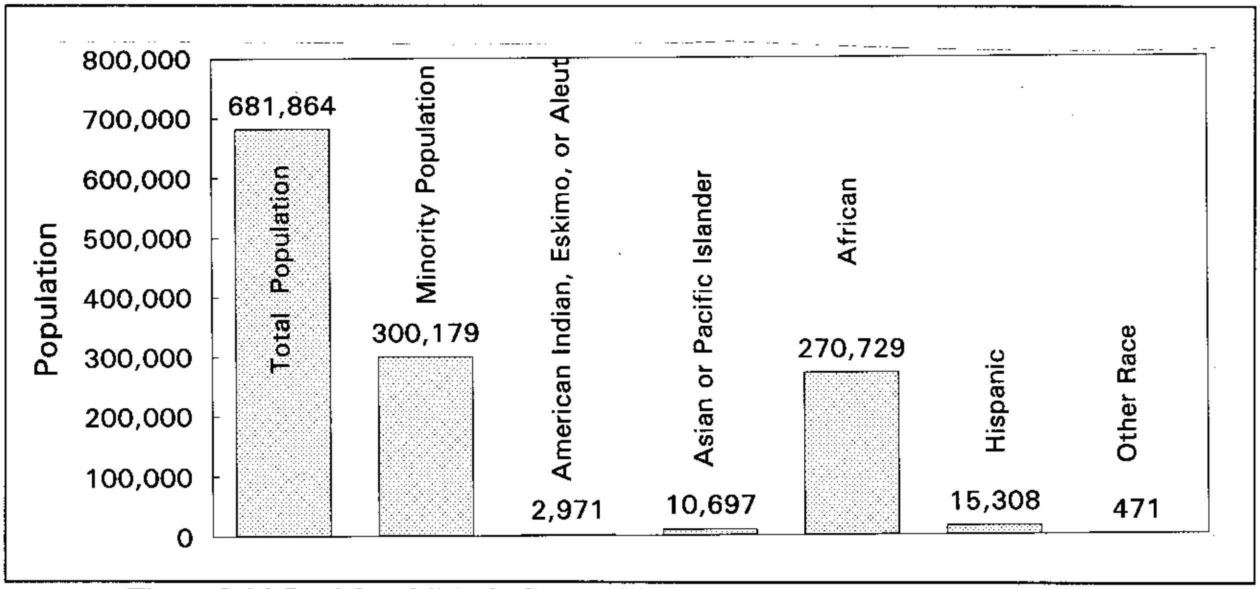


Figure 3-16 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Norfolk

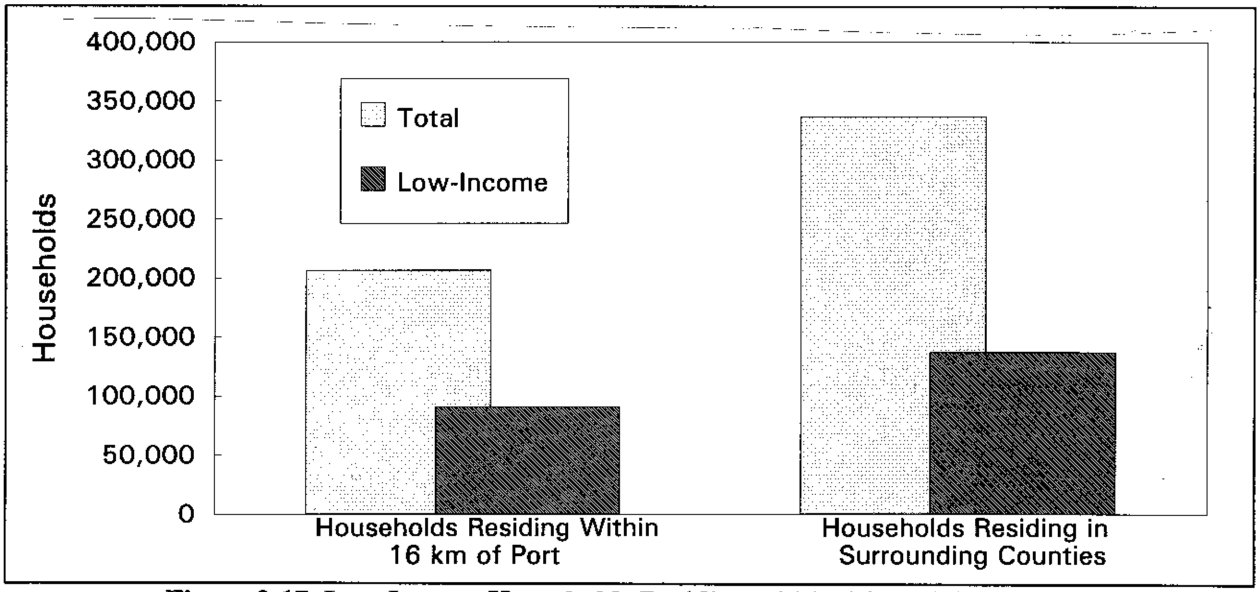


Figure 3-17 Low-Income Households Residing within 16 km (10 mi) of the Port of Norfolk

low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.4 Jacksonville, FL

The Port of Jacksonville is located on the Atlantic Coast of northern Florida, along the St. Johns River. It is a geographically large city (1,967 km^2 or 760 mi^2) ranging from the town of Orange on the east side of the river to Julington Creek on the west side. Most of the marine terminals are on the west side of the river, about 34 km (21 mi) from the ocean entrance. However, the Blount Island container terminal is well

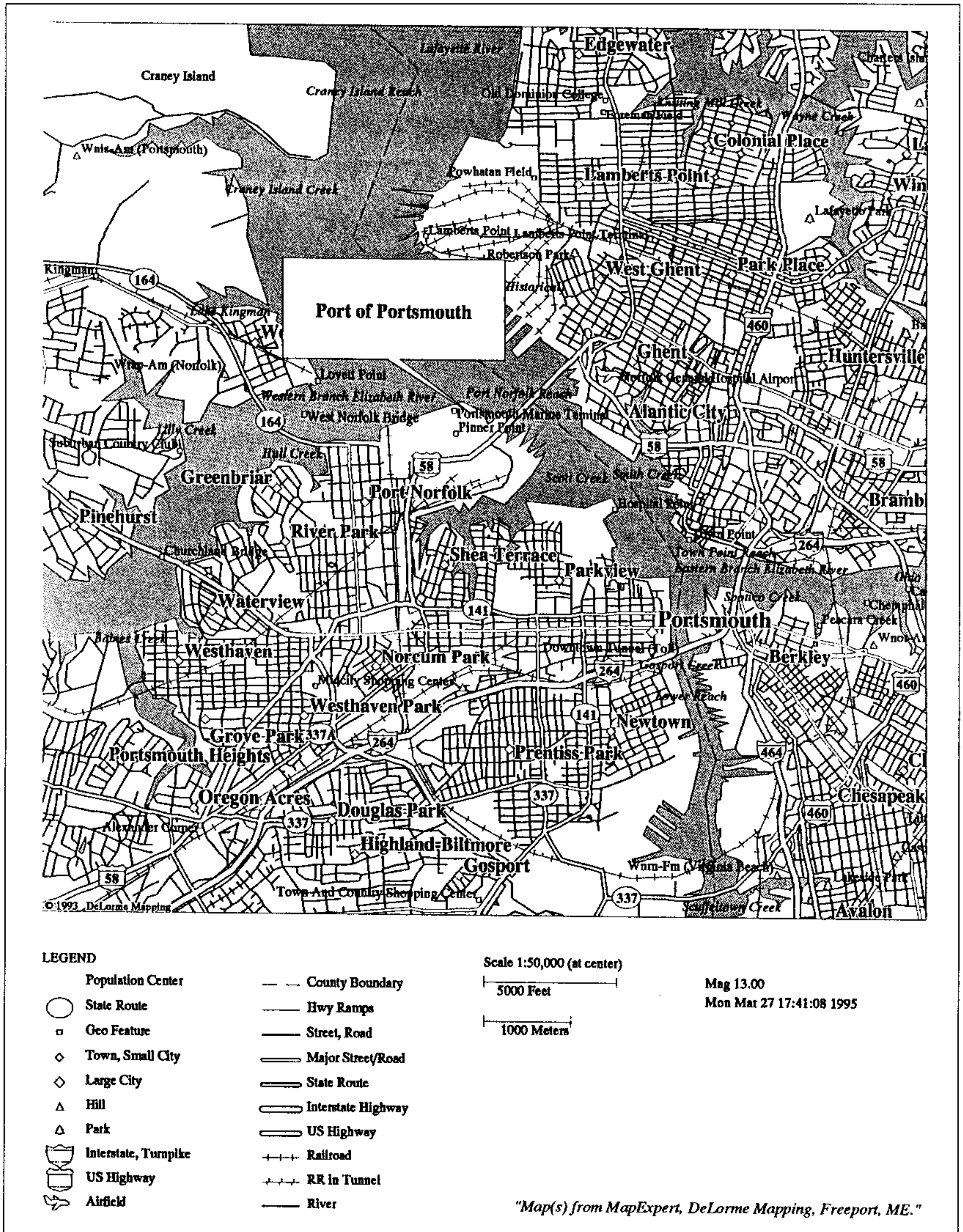


Figure 3-18 Port of Portsmouth, VA

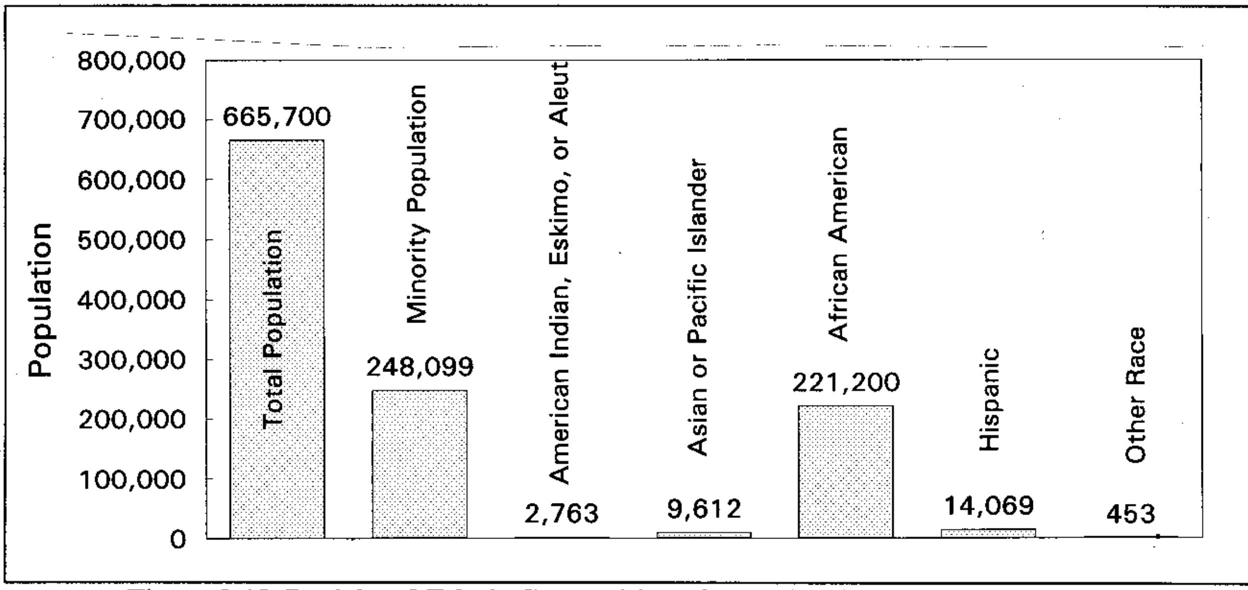


Figure 3-19 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Portsmouth

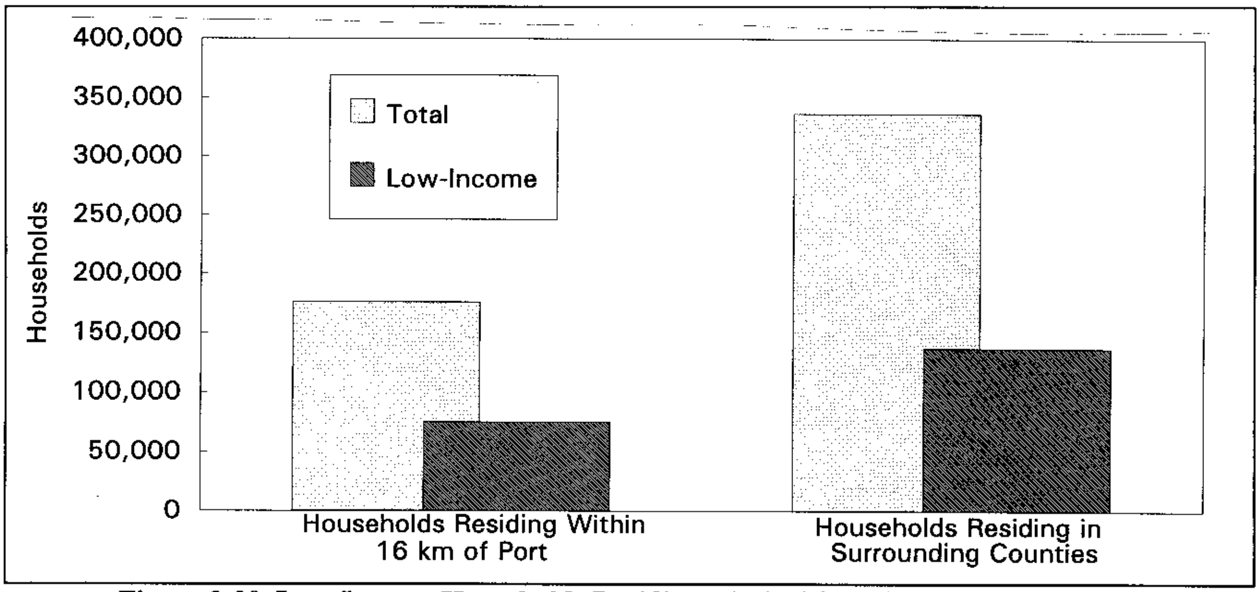


Figure 3-20 Low-Income Households Residing within 16 km (10 mi) of the Port of Portsmouth

separated from the city, and is only about 11 km (7 mi) from the harbor entrance. A map of the port is shown in Figure 3-21. A Federal project maintains a channel depth of 12.2 m (40 ft) to 12.8 m (42 ft) at the entrance to the river.

The St. Johns River has a deep, steep-sided channel cut through rock in some areas. Tidal currents are strong in the river as far as Jacksonville, approaching 3 knots in several places.

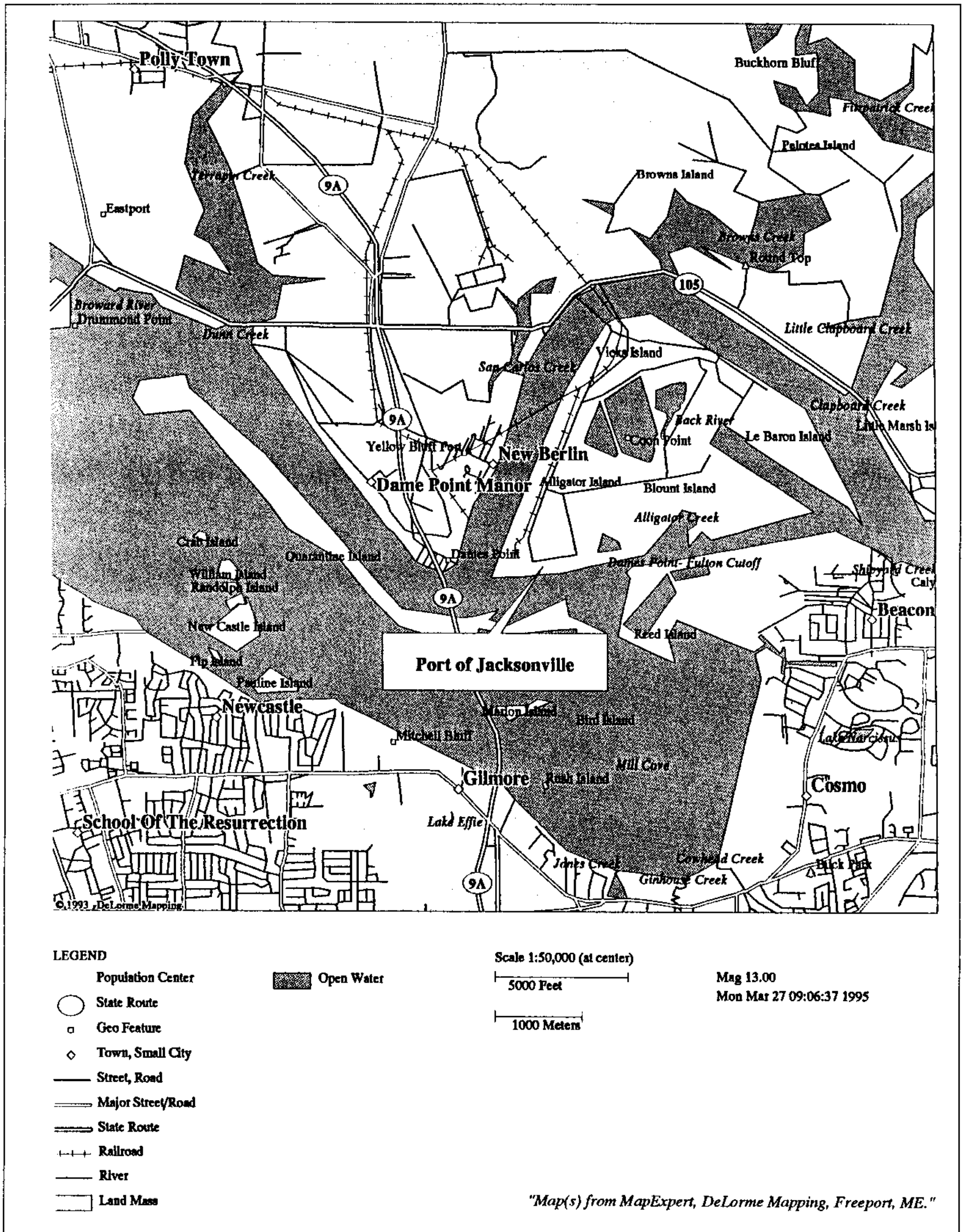


Figure 3-21 Port of Jacksonville, FL

There are two deepwater container/general cargo terminals: Blount Island, located approximately 11 km (7 mi) from the harbor entrance, and Talleyrand Docks and Terminals located about 34 km (21 mi) from the entrance. Both terminals are equipped with modern cranes, handle breakbulk and other types of cargo, and have warehouse as well as open storage areas. Of the two, Blount Island would be preferred because of its separation from the high-density downtown area and closer proximity to the sea.

Blount Island Terminal: Blount Island is a 356 ha (880 acre) facility with 1,920 m (6,336 ft) of berthing space. Blount Island berths 7-13 have 11.6 m (38 ft) of water alongside at mean low water, and five container cranes. This terminal is connected to the mainland via a fixed highway bridge which joins State Highway 105 (Necksher Drive) and connects with I-95 and Route 17 about five miles north of the city of Jacksonville. Blount Island has pierside service by the CSX Railroad, which connects with the Norfolk Southern Railroad.

Talleyrand Terminal: Talleyrand Docks is a 70 ha (173 acre) facility with 1,250 m (4,100 ft) of wharf on deep water (11.6 m or 38 ft at mean low water). It has two container cranes and two large gantry cranes. Talleyrand Terminal is located in downtown Jacksonville's shopping and commercial zone, about 2.9 km (1.8 mi) downstream of the John R. Matthews Bridge (alternate U.S. Route 90), and less than 1.0 km (0.6 mi) via city streets from the city Expressway system.

The 1990 population within 16 km (10 mi) of the port terminals was 334,212. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 46,900; the Oak Ridge Reservation, 175,000; the Idaho National Engineering Laboratory, 576,000; the Hanford Site, 643,000; and the Nevada Test Site, 639,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 607 km (377 mi); the Oak Ridge Reservation, 912 km (567 mi); the Idaho National Engineering Laboratory, 4,030 km (2,504 mi); the Hanford Site, 4,700 km (2,920 mi); and the Nevada Test Site, 4,190 km (2,604 mi). Distances along rail routes are about the same.

Environmental Conditions: The area between the mouth of the St. Johns River and Blount Island is characteristic of typical coastal lowlands found along the southeastern United States. Numerous creeks meander through large expanses of marshes and swamps. With the exception of the U.S. Naval Station Mayport and the village of Mayport, which occupy the first several miles along the southern bank of the river, the land bordering the lower portion of the river is largely undeveloped with the exception of riverfront residences, mainly along the northern bank. Most of the land to the north of the river between Blount Island and the coast is part of the Nassau River - St. Johns River Marshes Aquatic Preserve. The Fort Caroline National Memorial is located southeast of Blount Island on the southern bank of the river. The Little Talbot Island State Park is located approximately 1.6 km (1 mi) north of the channel entrance.

The lower 24.2 km (15 mi) of the St. Johns River has been designated as critical habitat for the manatee, a listed endangered species. The river is also used as a migratory area for the shortnose sturgeon, a listed endangered species (FWS, 1980b). According to the Florida Natural Areas Inventory, the following rare species have been reported within 3.2 km (2 mi) of the Blount Island Terminal: West Indian manatee (State and Federally Listed Endangered Species), shortnose sturgeon (State and Federally Listed Endangered Species), Atlantic sturgeon (State-Listed Species of Special Concern and Federally Listed Threatened Species), sea lamprey, and the opossum pipefish (Murray, 1994).

A variety of wading birds is also found in the vicinity of the Fort Caroline National Memorial. Several species of birds, including shorebirds, waterfowl, and gannets frequent the area around the jetties at the channel entrance. In particular, the brown pelican, a State Species of Special Concern, is found in this area. A variety of birds inhabits the Little Talbot Island State Park, including the American oystercatcher, a State Species of Special Concern. Loggerhead sea turtles, a listed endangered species, use the beaches along this portion of Florida as a nesting area (FWS, 1980b).

While environmental awareness is high throughout the state of Florida, there are no known sensitive wildlife sanctuaries in the immediate area of the Port of Jacksonville. Blount Island is surrounded by extensive marsh and wetlands.

Climatic Conditions: The Port of Jacksonville is located along the upper 39.4 km (24.5 mi) of the St. Johns River. The terrain in this area is relatively level, providing very little change in relief proceeding inland from the coastal region. The National Weather Service has been archiving meteorological information for this area since 1880.

The climate of this area is modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the Jacksonville area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months, becoming more southwesterly during spring and summer. Although Jacksonville is along the eastern U.S. coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years which have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving overland prior to reaching metropolitan Jacksonville. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992b). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Jacksonville, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Ethnic and Income Characteristics: Figure 3-22 shows the ethnic composition for the area surrounding the port at Jacksonville. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans were the largest minority group. Figure 3-23 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.5 Military Ocean Terminal Sunny Point, NC

The Military Ocean Terminal at Sunny Point (MOTSU) is a U.S. Department of Defense transportation facility located north of Southport, NC. The facility is located on the Cape Fear River, approximately 16 km (10 mi) upstream (north) from the mouth of the river, and 26 km (16.1 mi) south of the Port of Wilmington, NC. A map of MOTSU is shown in Figure 3-24. The port is easily accessed from the ocean, and all commercial vessels bound for Wilmington, NC must pass by MOTSU. It is served by a 12.1 m- (40 ft-) deep by 152 m- (500 ft-) wide channel from the ocean.

The water depth (channel and alongside the wharves) of 10.3 m (34 ft) at mean low water is adequate for most commercial breakbulk, roll-on/roll-off, and container ships. The terminal has three 600 m (2,000 ft) wharves, each with three berths. All wharves have three parallel sets of rail tracks. Berth 1, on the south

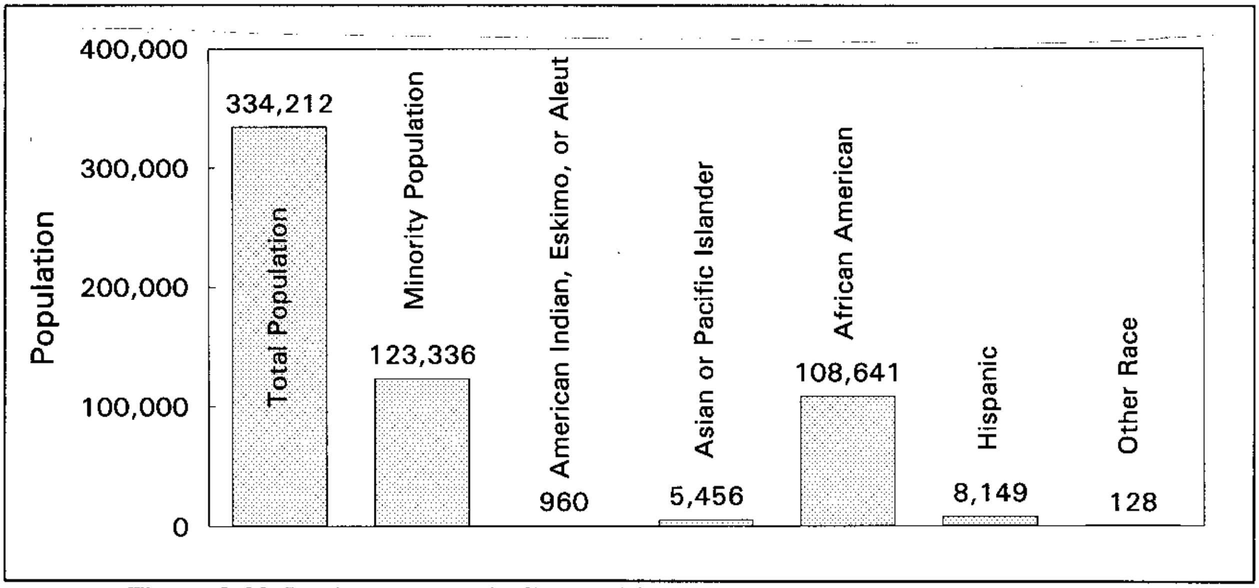


Figure 3-22 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Jacksonville

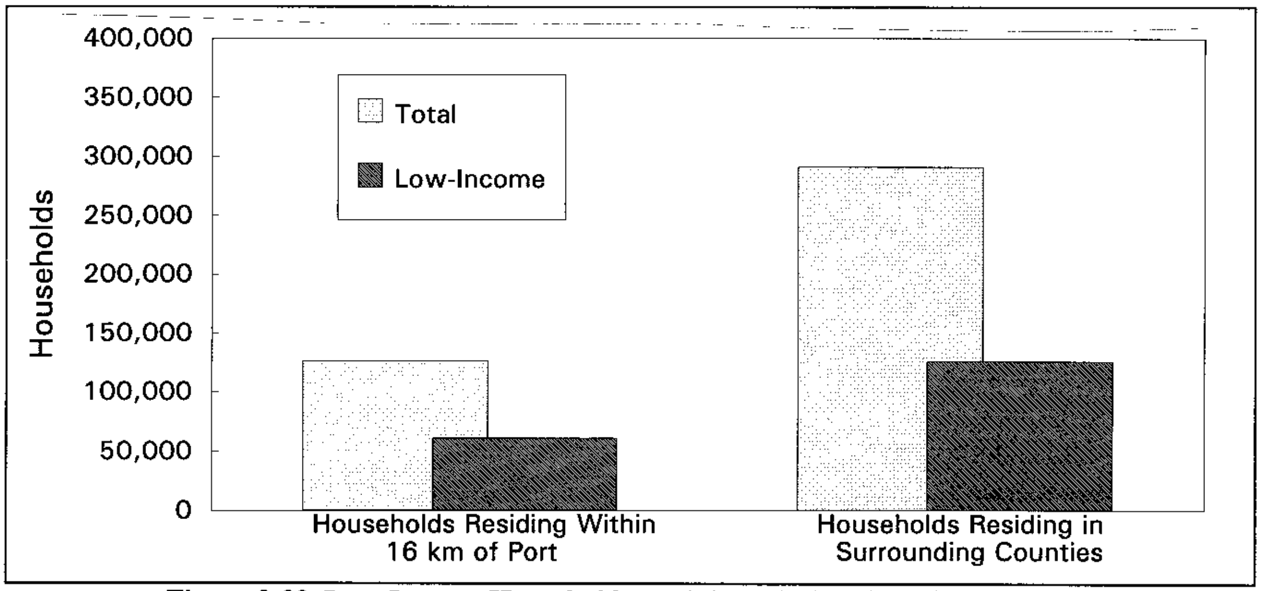


Figure 3-23 Low-Income Households Residing within 16 km (10 mi) of the Port of Jacksonville

wharf, has two 50 metric ton (55 ton) container cranes capable of off-loading container or container/breakbulk vessels. Berth 3 has been modified with a 30 m (100 ft) wide, reinforced concrete apron that permits breakbulk and roll-on/roll-off operations, in addition to containerized cargoes.

MOTSU is serviced by well-maintained roads, and has a dedicated 157 km (98 mi) U.S. Army rail line that connects the CSX network directly to the terminal. Truck access is provided by State Route 87 from the northwest and State Route 133 from the north. Route 87 provides access to U.S. 17, which runs southwest or northeast. The distance from the terminal gate to Route 133 is about 3.7 km (2.2 mi). Route 133 runs directly to U.S. 17 just outside Wilmington. From Wilmington, U.S. 74 runs west 120 km (75 mi) to Interstate 95, the nearest major north-south highway.

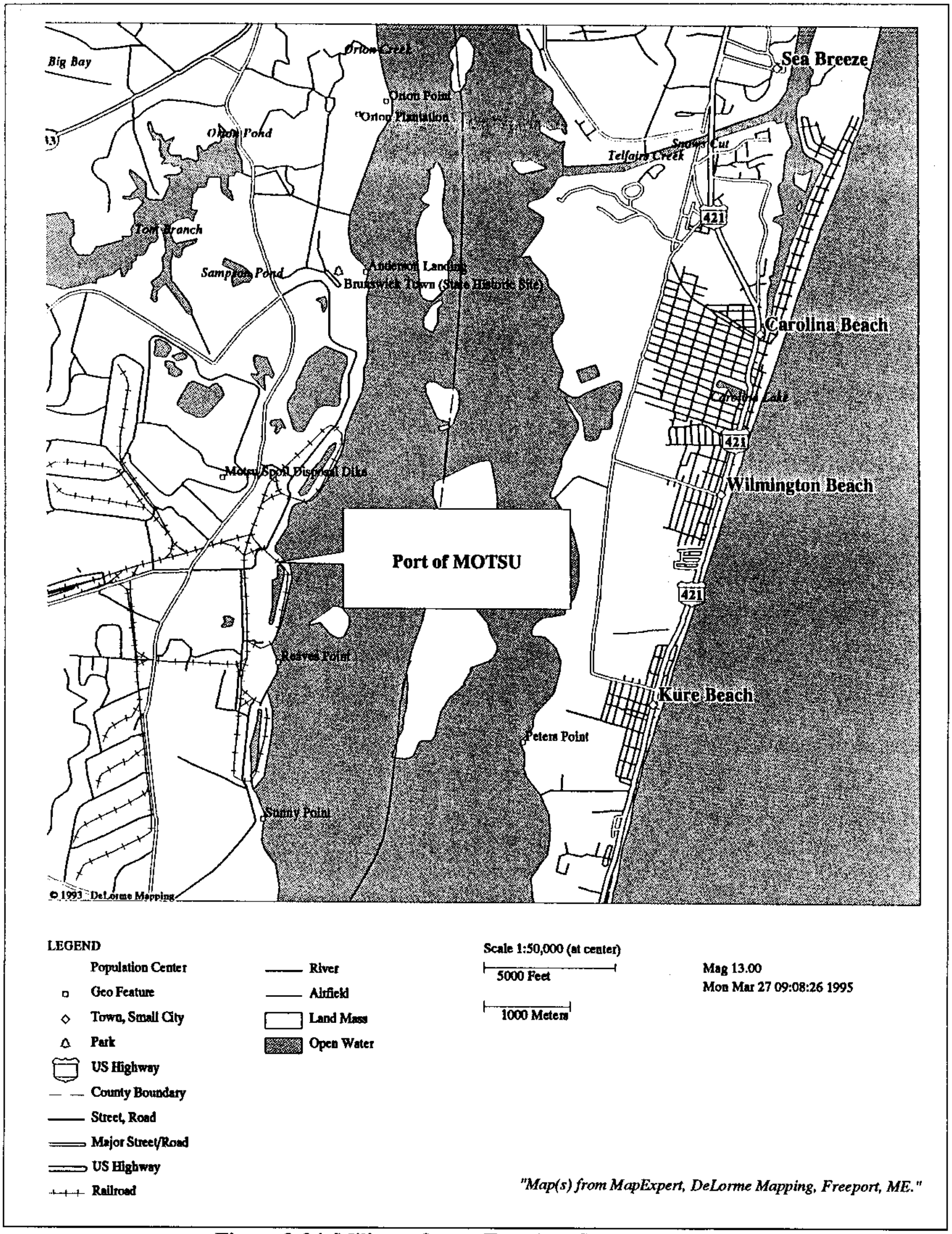


Figure 3-24 Military Ocean Terminal Sunny Point, NC

The 1990 population within 16 km (10 mi) of the port terminals was 7,995. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 34,200; the Oak Ridge Reservation, 128,000; the Idaho National Engineering Laboratory, 463,000; the Hanford Site, 548,000; and the Nevada Test Site, 619,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 402 km (250 mi); the Oak Ridge Reservation, 798 km (496 mi); the Idaho National Engineering Laboratory, 3,873 km (2,407 mi); the Hanford Site, 4,615 km (2,868 mi); and the Nevada Test Site, 3,953 km (2,456 mi). Distances along rail routes are slightly longer.

Environmental Conditions: The environmental conditions at MOTSU are similar to those at the Port of Wilmington, NC, and are described in Section 3.2.1.10. MOTSU has been identified as an area with sinkhole activities (Koch, 1984). Sinkholes can occur naturally or as a result of human activity. Occurrences of sinkholes are closely tied to the drainage pattern in areas where geologic formations provide a collapse mechanism. Human activities which modify the natural drainage pattern in such an area can increase the rate of sinkhole formation. Sinkholes pose a potential hazard to truck and rail traffic in the Sunny Point area. Due to the robust casks which would be used to transport spent nuclear fuel from foreign research reactors (see Appendix B for a description of the casks), sinkholes would not be expected to cause a radiological accident.

Climatic Conditions: The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For MOTSU, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a low seismic zone with an acceleration of 0.075 g.

Other climatic conditions at MOTSU are similar to those at the Port of Wilmington, NC, and are described in Section 3.2.1.10.

Ethnic and Income Characteristics: Figure 3-25 shows the ethnic composition for the area surrounding MOTSU. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 17 percent of the total population, and approximately 91 percent of the minority population for the area surrounding the port. Figure 3-26 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.6 Naval Weapons Station, Concord, CA

Naval Weapons Station (NWS) Concord is located on the southern edge of Suison Bay, an estuarine area immediately west of the junction of the Sacramento and San Joaquin Rivers. By water, the NWS is approximately 58 km (36 mi) northeast of the Golden Gate Bridge. The city of Concord, CA, is about 8 km (5 mi) south of the NWS. A map of the NWS military port is shown in Figure 3-27.

The 1990 population within 16 km (10 mi) of the port terminal was 381,070. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 1,040,000; the Oak Ridge Reservation, 742,000; the Idaho National Engineering Laboratory, 271,000; the Hanford Site, 263,000; and the Nevada Test Site, 437,000. Populations along rail routes to these sites are slightly smaller for the Oak Ridge Reservation, the Idaho National Engineering Laboratory and the Nevada Test Site, and slightly larger for the Savannah River Site and the Hanford Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,476 km (2,781 mi); the Oak Ridge Reservation, 4,111 km (2,554 mi); the Idaho National Engineering Laboratory,

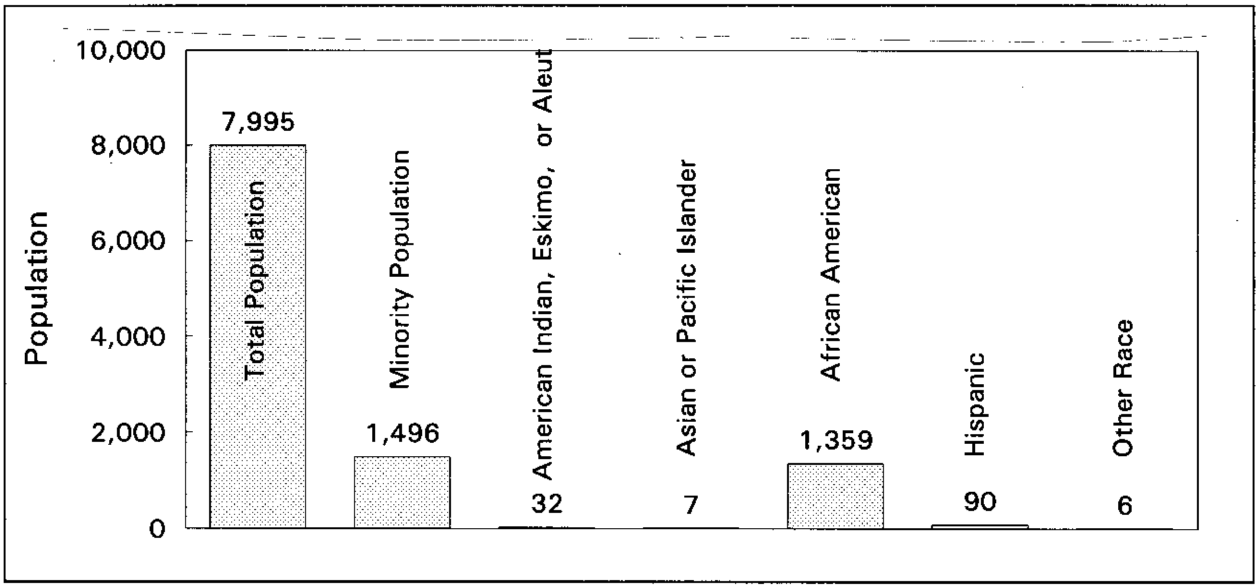


Figure 3-25 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of MOTSU

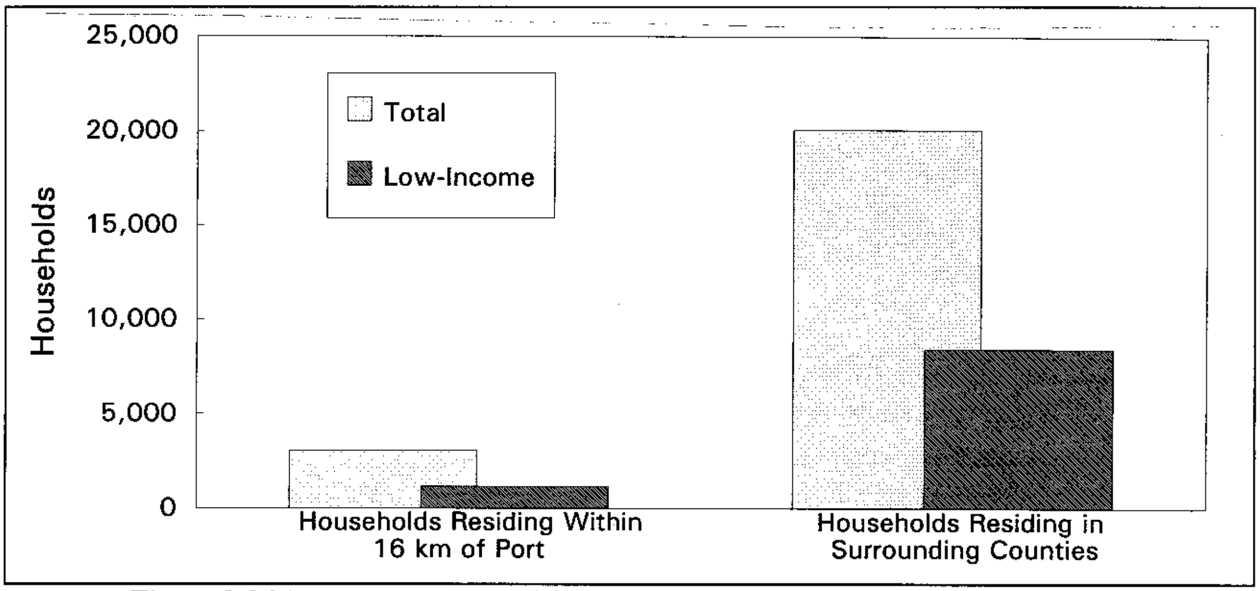
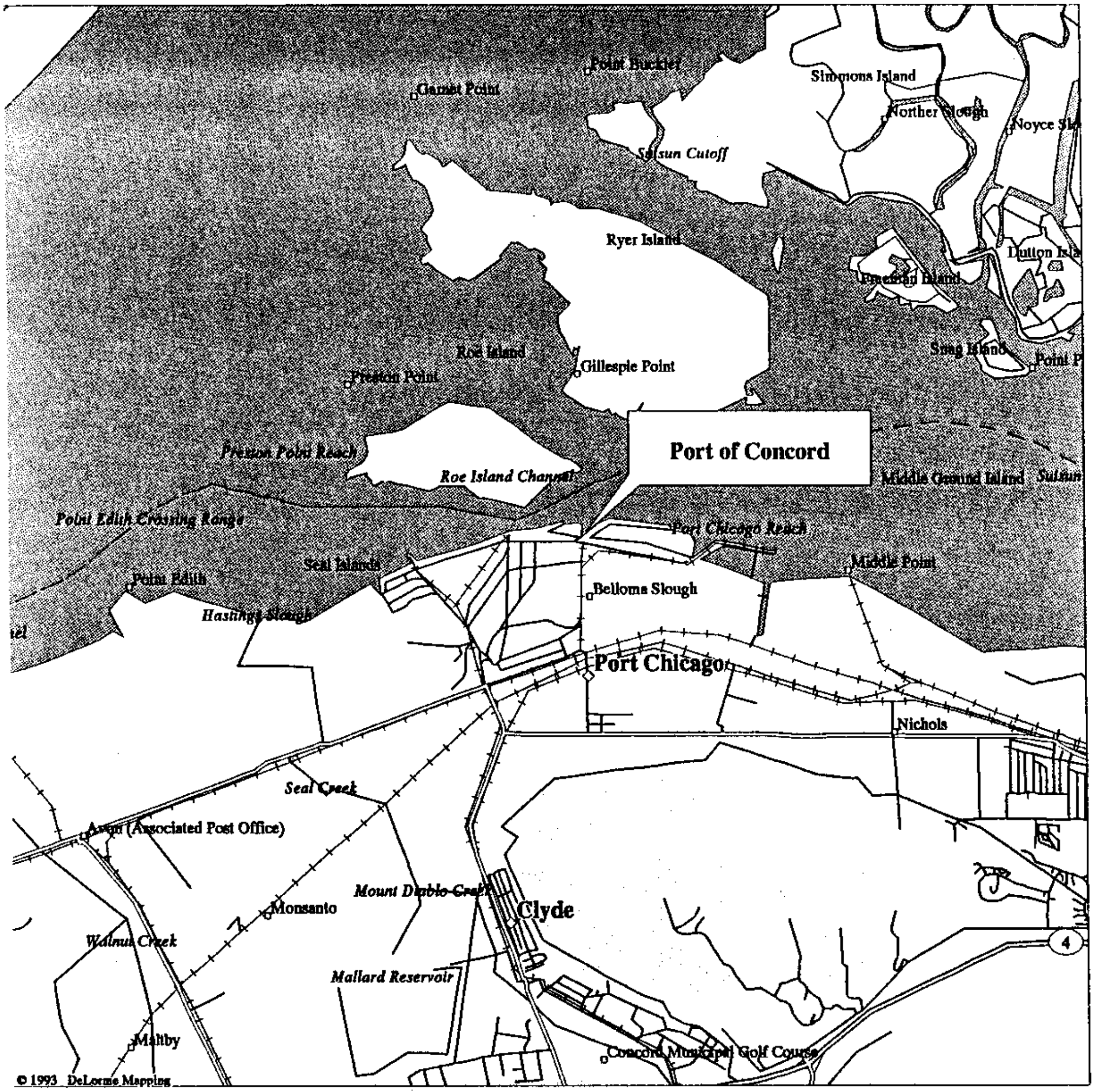


Figure 3-26 Low-Income Households Residing within 16 km (10 mi) of MOTSU

1,516 km (942 mi); the Hanford Site, 1,376 km (855 mi); and the Nevada Test Site, 1,145 km (711 mi). Distances along rail routes are about the same for the Idaho National Engineering Laboratory, and slightly longer for the Savannah River Site, the Oak Ridge Reservation, the Hanford Site, and the Nevada Test Site.

Environmental Conditions: NWS Concord occupies 5,233 ha (12,931 acres) of land adjoining south Suison Bay. Of this total acreage, 2,135 ha (5,276 acres) are inland, while 3,097 ha (7,653 acres) are more tidal in nature. Wetlands comprise approximately 1,215 ha (3,002 acres) of the tidal area (Yocum, 1994). Wetlands occupy large areas of land bordering all sides of Suison Bay and Grizzly Bay, which is located



LEGEND

- Population Center
- Geo Feature
- Town, Small City
- County Boundary
- Street, Road
- Major Street/Road
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00

Sun Mar 26 13:10:59 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure 3-27 Naval Weapons Station Concord, CA

directly north of Suison Bay. The waters of Suison Bay are characterized as a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). Chinook salmon, steelhead trout, striped bass, sturgeon, and American shad are typically found in this area (FWS, 1981a; FWS, 1981b).

Portions of the inland area at NWS Concord serve as a sanctuary for Tule elk, a formerly endangered species (Yocum, 1994). Other terrestrial species found in the area include the river otter, the salt-marsh harvest mouse (a Federally protected species), and the white-tailed kite. Adult concentrations and nesting areas of the California clapper rail (a Federally protected bird species) and the California black rail (a State protected species) are also found in this area. The Federally and State protected figwort plant family is also found in the vicinity of NWS Concord. In general, the greater San Francisco Bay area annually supports large numbers of shorebirds, wintering waterfowl, raptors, seabirds, and passerlings. In addition, shorebirds, wading birds, waterfowl, seabirds, and songbirds migrate through this coastal area.

Climatic Conditions: The climate is mild, with plenty of sunshine year round. Cloudless skies prevail during the spring, summer, and fall. Winter is the rainy season. Snow is rare, as are freezing temperatures. Sometimes torrential rains on the slopes can cause flooding along the Sacramento River (NOAA, 1993b).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Concord NWS, the Uniform Building Code provides a basic wind speed of about 110 km per hour (70 mi per hour). The port is located on the edge of a very high seismic zone with an acceleration of 0.45 g.

Ethnic and Income Characteristics: Figure 3-28 shows the ethnic composition for the area surrounding NWS Concord. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 7 percent of the total population, and approximately 24 percent of the minority population for the area surrounding the port. Other minorities include Asian or Pacific Islanders (11 percent of total population), and Hispanics (10 percent of total population). These groups constitute 38 percent and 35 percent of the minority population, respectively. Figure 3-29 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.7 Portland, OR

The Port of Portland is located about 160 km (100 mi) above the mouth of the Columbia River on the Willamette River tributary. Portland is the principal city of the Columbia River system, and one of the major ports on the Pacific Coast. The container terminal that would be used for potential receipt of foreign research reactor spent nuclear fuel is located approximately 170 km (106 mi) from the entrance of the Columbia River. Federal project depths in the Columbia River are 14.6 m (48 ft) at the mouth of the river, and 12 m (40 ft) at Portland. A map of the port is shown in Figure 3-30.

The 1990 census population within 16 km (10 mi) of the port was 356,064. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 686,000; the Oak Ridge Reservation, 519,000; the Idaho National Engineering Laboratory, 143,000; the Hanford Site, 85,700; and the Nevada Test Site, 375,000. Populations along rail routes to these sites are slightly smaller for the Nevada Test Site and the Idaho National Engineering Laboratory, but slightly larger for the Savannah River Site, the Oak Ridge Reservation, and the Hanford

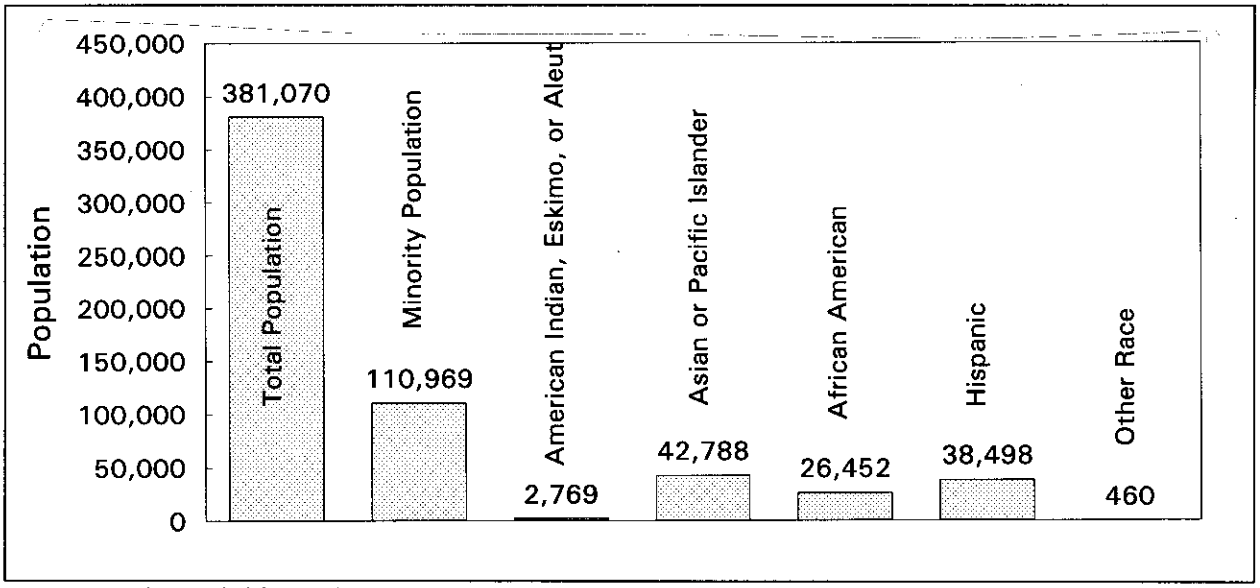


Figure 3-28 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of NWS Concord

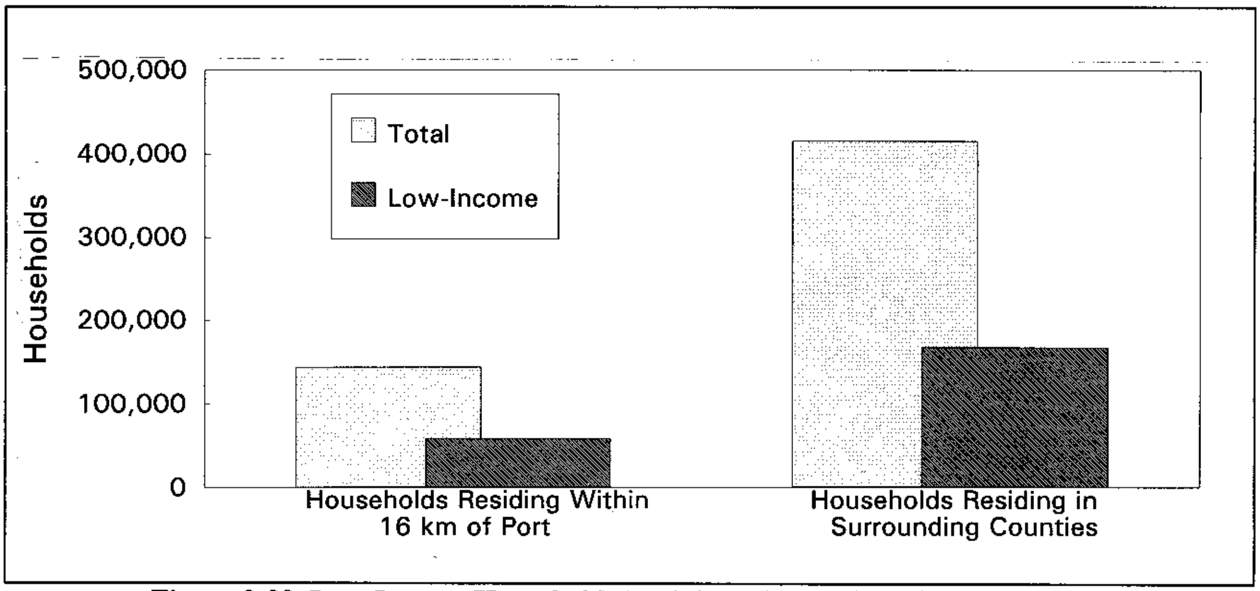


Figure 3-29 Low-Income Households Residing within 16 km (10 mi) of NWS Concord

Site. The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,630 km (2,877 mi); the Oak Ridge Reservation, 4,200 km (2,610 mi); the Idaho National Engineering Laboratory, 1,190 km (739 mi); the Hanford Site, 407 km (253 mi); and the Nevada Test Site, 2,040 km (1,268 mi). Distances along rail routes are slightly longer, with the exception of the Hanford Site, which is slightly shorter.

Environmental Conditions: There are no known areas of special environmental concern in the immediate vicinity of the port, although concern for the environment runs high throughout the Pacific Northwest. The areas surrounding the Terminal are in river-oriented industrial land use. Wildlife habitat along the

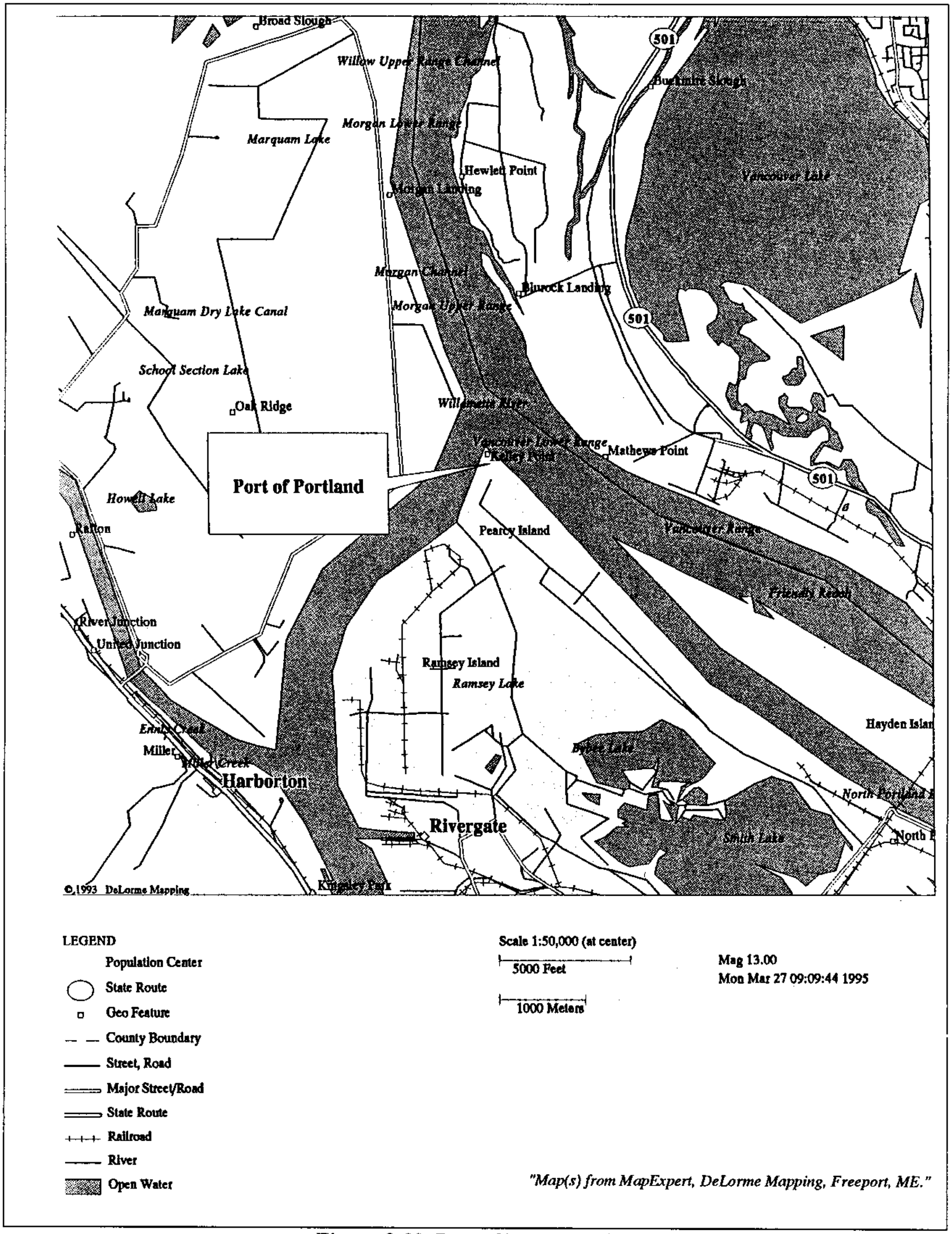


Figure 3-30 Port of Portland, OR

Oregon Slough is limited because of the industrial development, although some waterfowl use the area. While the primary uses in the Terminal area are commercial navigation and industry, some recreational fishing and boating occurs in Oregon Slough and the Columbia River.

The U.S. Fish and Wildlife Service's Ecological Inventory for the Vancouver, Washington-Oregon area indicates that the Columbia River generally includes the following fish species: salmonids, chinook salmon, coho salmon, chum salmon, pink salmon, sockeye salmon, steelhead trout, Dolly Varden, smelts, river lamprey, white sturgeon, American shad, eulachon and cutthroat trout (FWS, 1981c). South of Portland, the various islands and wetlands along the Columbia River provide habitat for a wide variety of terrestrial organisms. Areas of special interest include the Sauvie Island Game Management Area, which is located approximately 8 km (5 mi) downriver of Terminal 6, and the Ridgefield National Wildlife Refuge, which is approximately 16 km (10 mi) downriver.

The U.S. Army Corps of Engineers reports that raptors such as the red-tail hawk, bald eagle, and peregrine falcon are occasional visitors to this area and the U.S. Fish and Wildlife Service has indicated that the endangered American peregrine falcon and threatened bald eagle may winter in this area. In addition, the National Marine Fisheries Service has listed the Snake River sockeye salmon as endangered, and two Snake River chinooks stocks as threatened (Kurkoski, 1994). The State of Oregon's Natural Heritage Program reports that there are at least two rare species that occur in the vicinity of Terminal No. 6 (Gaines, 1994). These species are the painted turtle (a State-Sensitive-Critical species) and the Columbia water-meal.

Climatic Conditions: The port area is subject to earthquakes and volcanism (NOAA, 1992d). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. For the Port of Portland, the Uniform Building Code provides a basic wind speed of about 140 km per hour (90 mph) (UBC, 1991). The port is located in a moderate seismic zone with an acceleration of 0.20 g. There have been two major earthquakes in the Puget Sound area this century (Bolt, 1978). On May 18, 1980, nearby Mount St. Helens suffered a major volcanic eruption. All the mountains along the Cascade Range are volcanic in origin and have some potential for eruption.

Ethnic and Income Characteristics: Figure 3-31 shows the ethnic composition for the area surrounding the port at Portland. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 8 percent of the total population, and approximately 50 percent of the minority population for the area surrounding the port. Hispanics and Asian or Pacific Islanders each accounted for about 20 percent of the minority population. Figure 3-32 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.8 Savannah, GA

The Port of Savannah is located on the south bank of the Savannah River, about 35 km (22 mi) above the entrance from the Atlantic Ocean. Savannah is the third largest city in Georgia, and is the chief port of the State of Georgia. The Savannah River serves as the boundary between Georgia and South Carolina. There are three large cargo terminals at the port. One of these terminals, Containerport, is a dedicated container handling terminal. It is located about 45.6 km (28.3 mi) up the Savannah River from the Atlantic Ocean. A map of the port is shown in Figure 3-33.

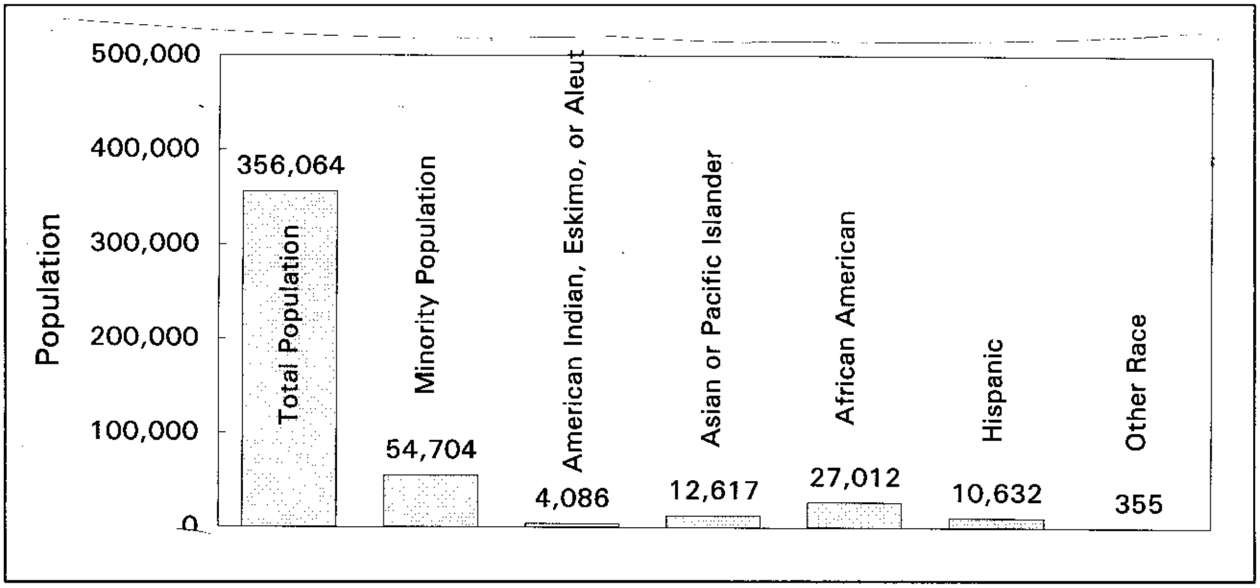


Figure 3-31 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Portland

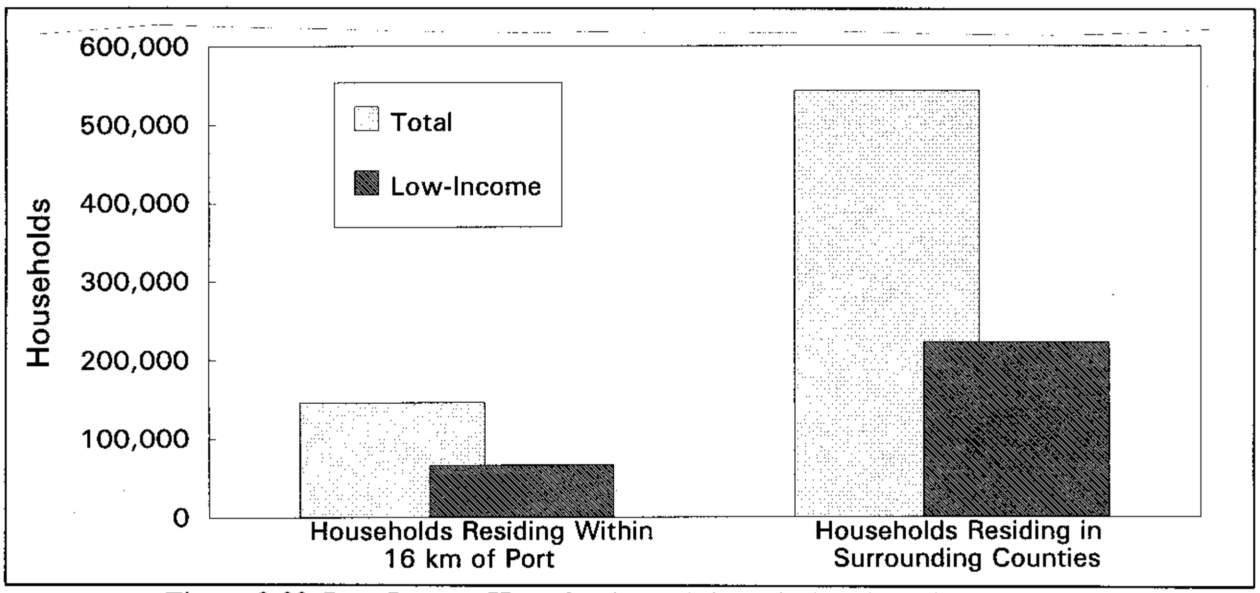


Figure 3-32 Low-Income Households Residing within 16 km (10 mi) of the Port of Portland

The 1990 population within 16 km (10 mi) of the port terminals was 155,166. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 37,300; the Oak Ridge Reservation, 101,000; the Idaho National Engineering Laboratory, 553,000; the Hanford Site, 602,000; and the Nevada Test Site, 616,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 400 km (249 mi); the Oak Ridge Reservation, 720 km (447 mi); the Idaho National Engineering Laboratory, 3,860 km (2,398 mi); the Hanford Site, 4,530 km (2,815 mi); and the Nevada Test Site, 4,020 km (2,498 mi). Distances along rail routes are slightly longer.

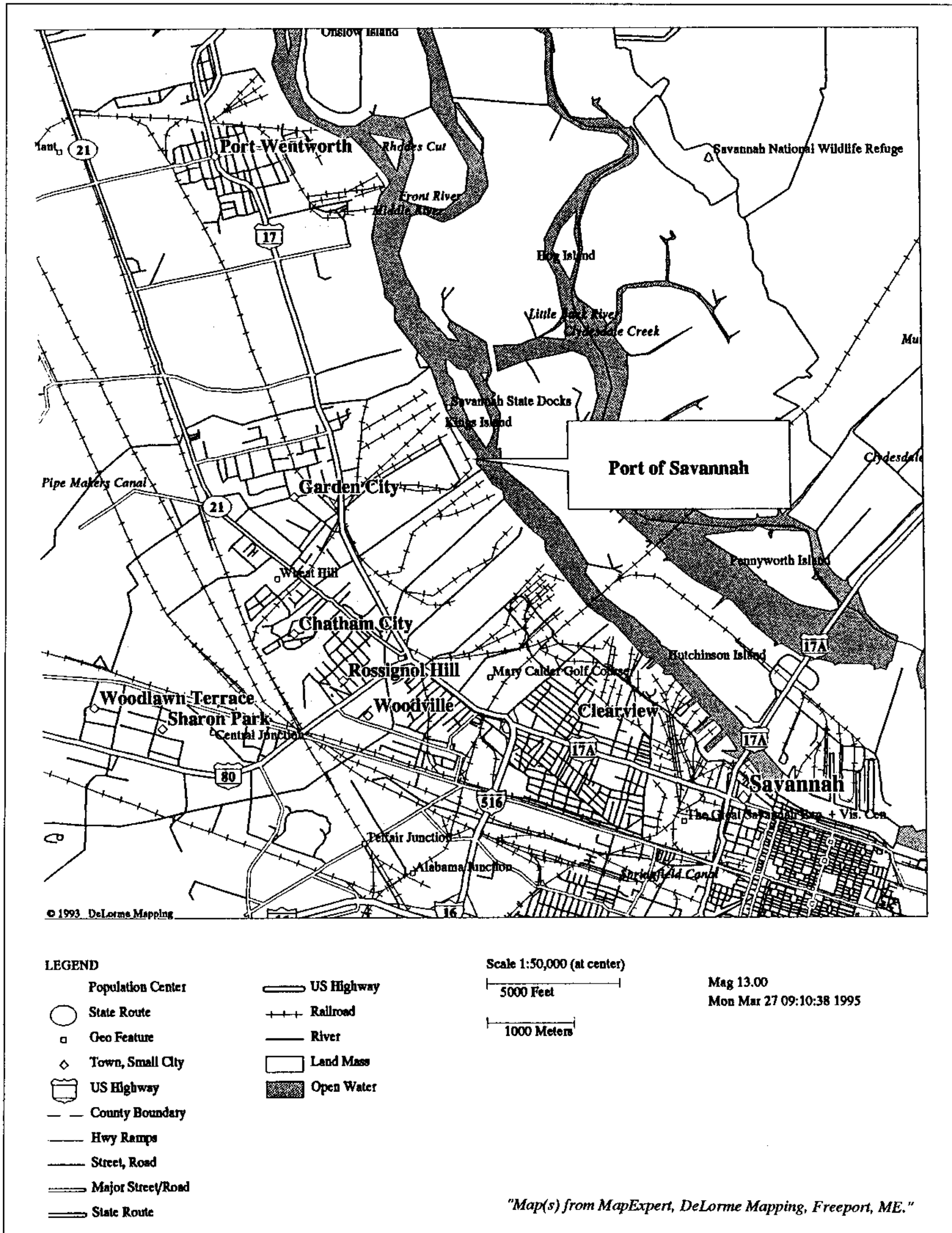


Figure 3-33 Port of Savannah, GA

Environmental Conditions: The lower Savannah River has multiple branches that meander through a variety of coastal lowlands including salt marshes, tidal creeks, freshwater marshes, and freshwater impoundments. South Carolina has classified its portion of the Savannah harbor upstream from Fort Pulaski (located at the mouth of the Savannah River) as Class B, and the portion oceanward as Class SA. Class B waters are freshwaters suitable for secondary contact recreation, use as a drinking water source following conventional treatment, fishing, industrial, and agricultural use. Class SA waters are defined as tidal salt waters suitable for primary contact recreation, and for all the uses listed in Class B. The State of Georgia has classified the Savannah River from mile 0 at Fort Pulaski north to mile 5 at Field's Cut as recreation waters. North of Field's Cut, the waters are classified as Coastal Fishing (U.S. Army, 1991). The river in the vicinity of Containerport is characterized as a transitional estuarine habitat, where the salinity ranges from low (generally 0.5 to 5 ppt) to mid-salinity (generally 5 to 16.5 ppt) (FWS 1980c).

A large number of aquatic and terrestrial species are found in and around the Savannah River near Containerport. State or Federally protected, endangered, or threatened aquatic species in the vicinity of Containerport include the Shortnose sturgeon and the Florida manatee, both identified as State and Federally endangered species. The Shortnose sturgeon uses the Savannah River as a migratory area. In addition, the Loggerhead turtle, the Bald eagle, and the American alligator are found along the lower reaches of the Savannah River (FWS, 1980c).

Both invertebrate and fish species of commercial and recreational value are found in the Savannah River. Commercial fishing is primarily for the American shad, sturgeon, shrimp, and blue crab. Public shellfishing is allowed in some areas near the mouth of the Savannah River, in the vicinity of Fort Pulaski. The Savannah River hosts the migration of several important commercial and game fishes, including the American shad, the hickory shad, and the blueback herring. Game species include the spotted seatrout, red drum, croaker, spot, striped bass, flounder, silver perch, white catfish, channel catfish, largemouth bass, sunfish, and crappies. The State of Georgia has closed the striped bass fishery for population recovery purposes (Schmitt, 1993).

There are several wildlife refuges and/or game management areas located along the lower portion of the Savannah River. Tybee National Wildlife Refuge is located at the mouth of the Savannah River at the confluence with the Atlantic Ocean. Just north of Tybee National Wildlife refuge is the Turtle Island Game Management Area. Containerport itself is located across the river from the southern end of the 10,371 ha (25,627 acre) Savannah National Wildlife Refuge. The Savannah National Wildlife Refuge and the Tybee National Wildlife Refuge are managed by the U.S. Fish and Wildlife Service.

Climatic Conditions: The area has a temperate climate, which is greatly influenced by winds coming into the area off the ocean. Nominally, 50 percent of the rainfall occurs during thunderstorms, with the remainder being equally distributed over the year and generally related to weather front passages. Severe tropical systems affect the Savannah area roughly once every 10 years and cause heavy, sustained precipitation, high winds, and extreme, but usually localized, coastal flooding. Rainfall measurements in excess of 51 cm (20 in) have been observed as a result of tropical systems impacting the area. Based on the 1951-1980 climatology, the first freeze occurs on average around November 15, and the last near March 10 (NOAA, 1992e).

The Port is subject to severe hurricanes and tropical storms, and given its proximity to Charleston, SC may have a slightly higher risk of earthquakes than the rest of the State of Georgia. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Savannah, the Uniform Building code provides a basic wind speed of about 130 km per hour (80 mi per hour). The port is located in a low seismic zone with an acceleration of 0.075 g.

Ethnic and Income Characteristics: Figure 3-34 shows the ethnic composition for the area surrounding the port at Savannah. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 49 percent of the total population, and approximately 95 percent of the minority population for the area surrounding the port. Figure 3-35 shows analogous information for low-income households residing within 16 km of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

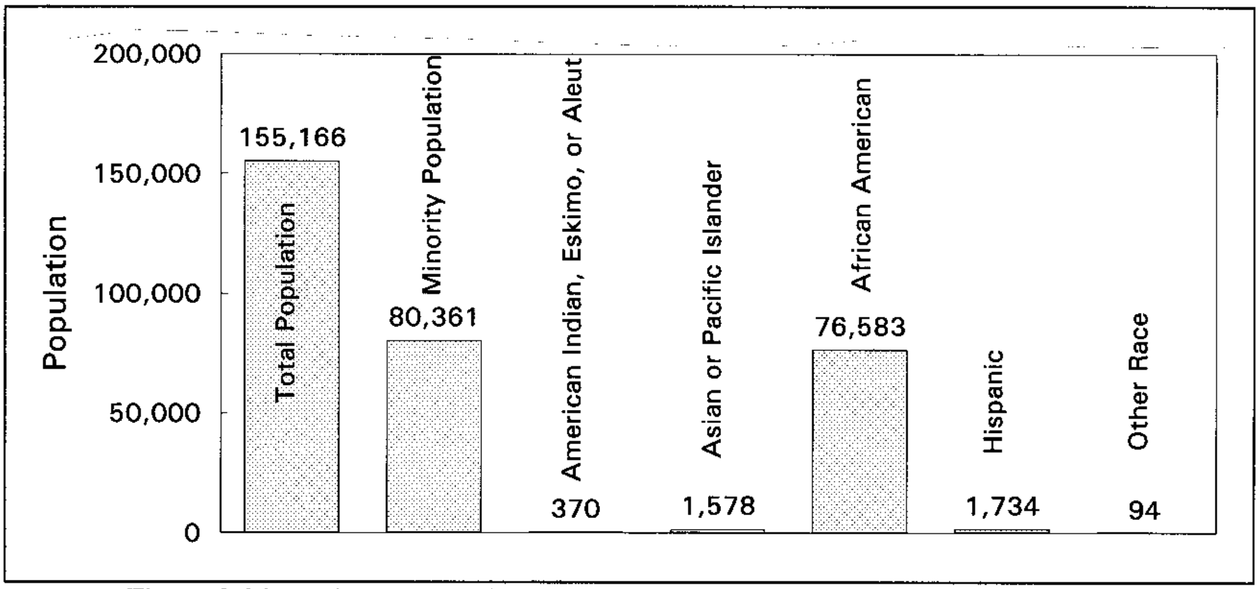


Figure 3-34 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Savannah

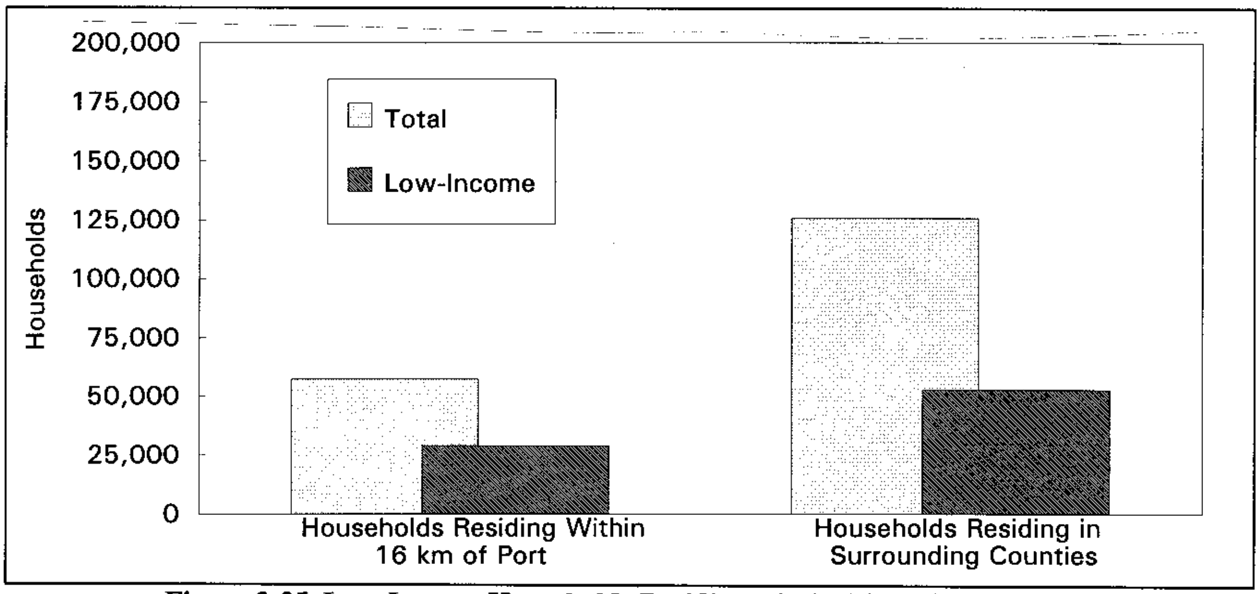


Figure 3-35 Low-Income Households Residing within 16 km (10 mi) of the Port of Savannah

3.2.1.9 Tacoma, WA

The Port of Tacoma is located in the southeastern corner of Puget Sound on the deep waters of Commencement Bay about 5 km (3 mi) from the Sound. It is a rapidly expanding major port second only to Seattle in maritime importance on Puget Sound. The distance from the entrance into Puget Sound is approximately 130 km (80 mi). A map of the port is shown in Figure 3-36.

Terminal 7, Berth D is the primary container terminal. It has one 274 m (904 ft) long container berth, 3 container cranes, and 15.2 m (50 ft) of depth alongside at mean low water.

The terminal is about 4.8 km (3 mi) from the Port of Tacoma road access to Interstate 5 immediately outside the port complex. A somewhat longer route, Interstate 5 South, connects with I-84 East near Portland, OR. Ship berths are served by the Port Belt Line Railroad, and the port is served by the Burlington Northern and Union Pacific Railroads, which interline with eastern and southern railroads.

The 1990 population within 16 km (10 mi) of the port was 511,575. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 601,000; the Oak Ridge Reservation, 431,000; the Idaho National Engineering Laboratory, 157,000; the Hanford Site, 98,600; and the Nevada Test Site, 379,000. Populations along rail routes to four of these sites are slightly larger, but the population along the rail route to the Nevada Test Site is slightly smaller (this is largely due to primary use of interstate highways through Salt Lake City, UT and Las Vegas, NV). The distances to the five potential sites on interstate routes are: the Savannah River Site, 4,720 km (2,933 mi); the Oak Ridge Reservation, 4,280 km (2,659 mi); the Idaho National Engineering Laboratory, 1,310 km (814 mi); the Hanford Site, 399 km (248 mi); and the Nevada Test Site, 2,160 km (1,342 mi). Distances along rail routes are much longer.

Environmental Conditions: A variety of marine mammals can be found in central Puget Sound including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the gray whale, the humpback whale, the Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995c), although these species are not reported at the port. Bald eagles can be found throughout this coastal zone, and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The U.S. Fish and Wildlife Services' Ecological Inventory for the Puget Sound area indicates that the habitat of Commencement Bay is used by a variety of birds including shorebirds, gulls, sandpipers, turnstones, yellowlegs, herons, rails, great blue herons, waterfowl, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American wigeons, pintails, mallards, seabirds, cormorants, alcids, common murres, and pigeon guillemots. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of chinook salmon, coho salmon, chum salmon, and pink salmon are found in the Puyallup Waterway/River and use this water body and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, a number of seabird colonies exist along the shoreline of Commencement Bay. Areas of the Puget Sound, north of Commencement Bay, are also used as haulouts by the California Sea Lion. Areas of estuarine wetlands are located along the northern shore of Commencement Bay (WDW, 1994).

Climatic Conditions: The mild climate of the Pacific Coast is modified by the Cascade Mountains and to a lesser extent by the Olympic Mountains. The climate is characterized by mild temperatures, a well-defined rainy season and prolonged cloud cover, especially during the winter months. The Cascades act as a very effective barrier in both winter and summer, shielding the region from both extreme cold and

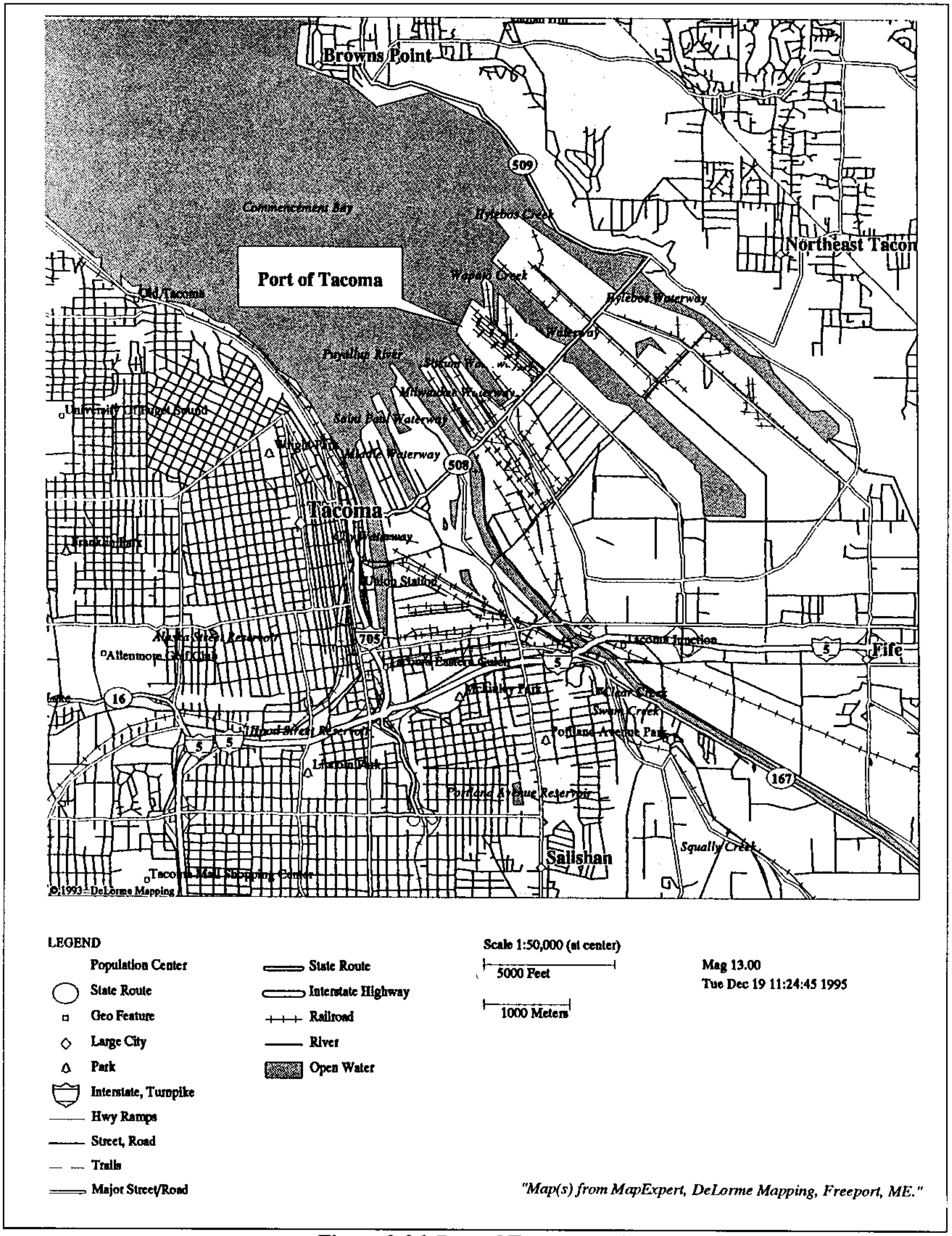


Figure 3-36 Port of Tacoma, WA

heat, respectively. The rainy season extends from October through March, with December accounting for the most rainfall. Approximately 75 percent of the annual total precipitation occurs during the winter rainy season. The dry season is centered around July and August. The majority of Seattle's precipitation is associated with normal, mid-latitude disturbances, which are most vigorous during the winter months. During summer, the dominant storm track (e.g., the polar jet) shifts northward into southern Canada, reducing the precipitation in the area. Summer thunderstorms do occur but do not contribute measurably to the annual rainfall budget. Prevailing winds are from the southwest, but occasional severe winter storms will produce strong northerly winds. Summer winds are generally rather light, with the occasional evidence of land-sea breeze effects creating northerly flows. Fog and low-level stratocumulus clouds form over the southern Puget Sound area in the late summer, fall, and early winter months, and often dominate the weather conditions of the early morning hours, reducing surface visibility. Based on 1951-1980 climatology, the first occurrence of freezing temperatures should occur around November 11, and the last incidence in spring around March 24 (NOAA, 1992f).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code. For the Port of Tacoma, the Uniform Building Code provides a basic wind speed of about 130 km per hour (80 mph) (UBC, 1991). The port is located in a high seismic zone with an acceleration of 0.30 g. There have been two major earthquakes in the Puget Sound area this century (Bolt, 1978). On May 18, 1980, Mount St. Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascade Range from Canada to Northern California are volcanic in origin and are potentially active.

Ethnic and Income Characteristics: Figure 3-37 shows the ethnic composition for the area surrounding the port at Tacoma, WA. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans constituted the largest minority group at about 6 percent of the total population, and approximately 38 percent of the minority population for the area surrounding the port. Asian and Hispanic minorities make up approximately 33 percent and 20 percent, respectively, of the minority population. Native Americans make up about 8 percent of the minority population near the port of Tacoma. Figure 3-38 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

3.2.1.10 Wilmington, NC

The Port of Wilmington, NC, is located on the east bank of the Cape Fear River, about 42 km (26 mi) above its mouth. It is the leading port of North Carolina, and its major export is wood pulp. The major terminals are down river from the city. A Federal project maintains a 12.2 m (40 ft) channel at the mouth of the Cape Fear River, 11.6 m (38 ft) to the port. A new dredging program will deepen the approach channel to 12.2 m (40 ft). A map of the port is shown in Figure 3-39.

The Wilmington wharves are of concrete pile construction, rubber fendered, with a total frontage of about 2,000 m (6,600 ft). Berths 6 to 9 are dedicated containership berths, with the remaining berths used for various kinds of general cargo. All of the main cargo berths have a depth alongside of 11.6 m (38 ft) at mean low water. The terminal has five container cranes, plus three gantry cranes (Jane's, 1992; AAPA, 1993; FHI, 1993).

Truck shipments from the port to southern destinations are along U.S. Routes 17, 74, 76, and 421 to Interstates 95 and 40 (POW, 1994). Northern and western long-distance routes are via Interstate 40, which connects with State Highway 132 about 16 km (10 mi) north of the city. The 1990 population within

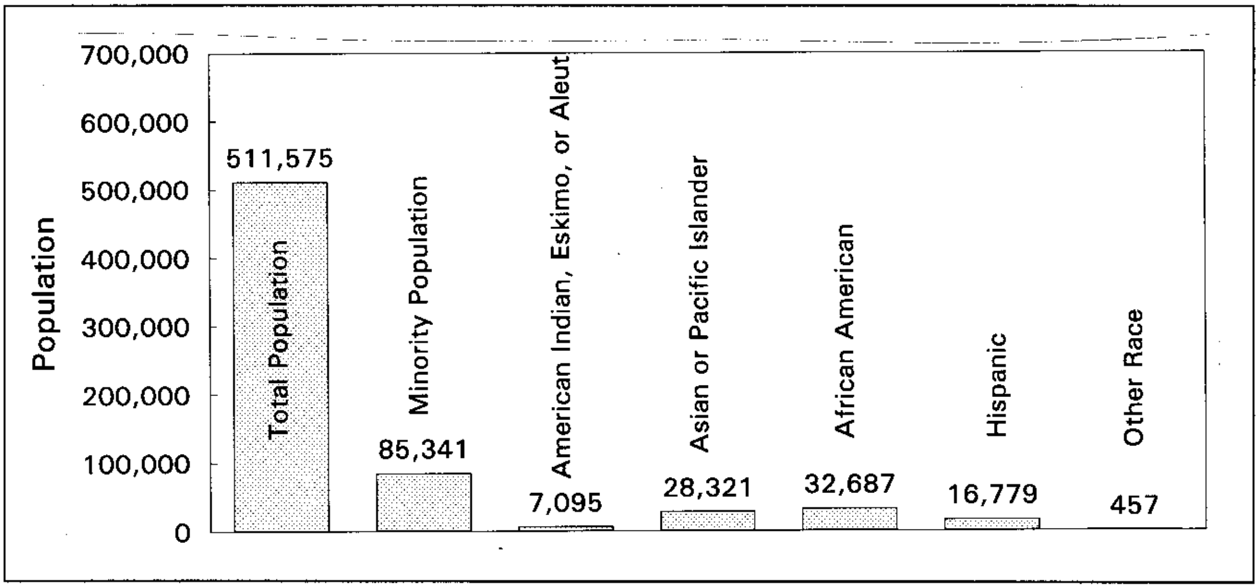


Figure 3-37 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Tacoma

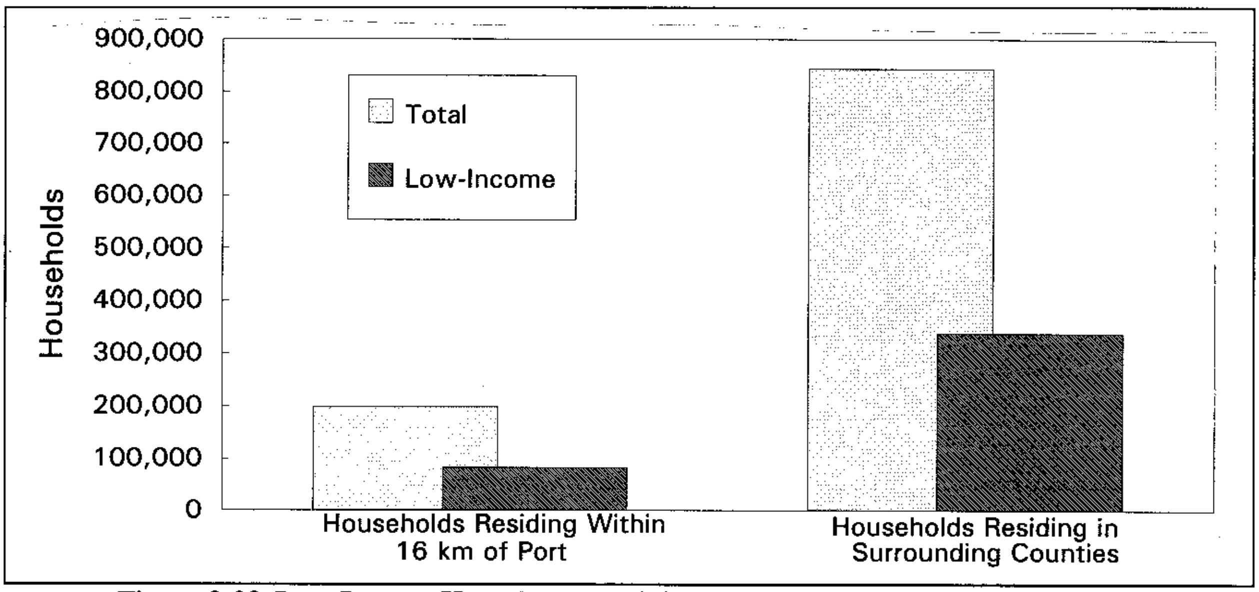


Figure 3-38 Low-Income Households Residing within 16 km (10 mi) of the Port of Tacoma

16 km (10 mi) of the port terminals was 115,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: the Savannah River Site, 64,700; the Oak Ridge Reservation, 128,000; the Idaho National Engineering Laboratory, 507,000; the Hanford Site, 556,000; and the Nevada Test Site, 570,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: the Savannah River Site, 500 km (311 mi); the Oak Ridge Reservation, 820 km (510 mi); the Idaho National Engineering Laboratory, 4,100 km (2,548 mi); the Hanford Site, 4,770 km (2,964 mi); and the Nevada Test Site, 4,260 km (2,647 mi). Distances along rail routes are slightly longer for western sites, but about the same for eastern sites.

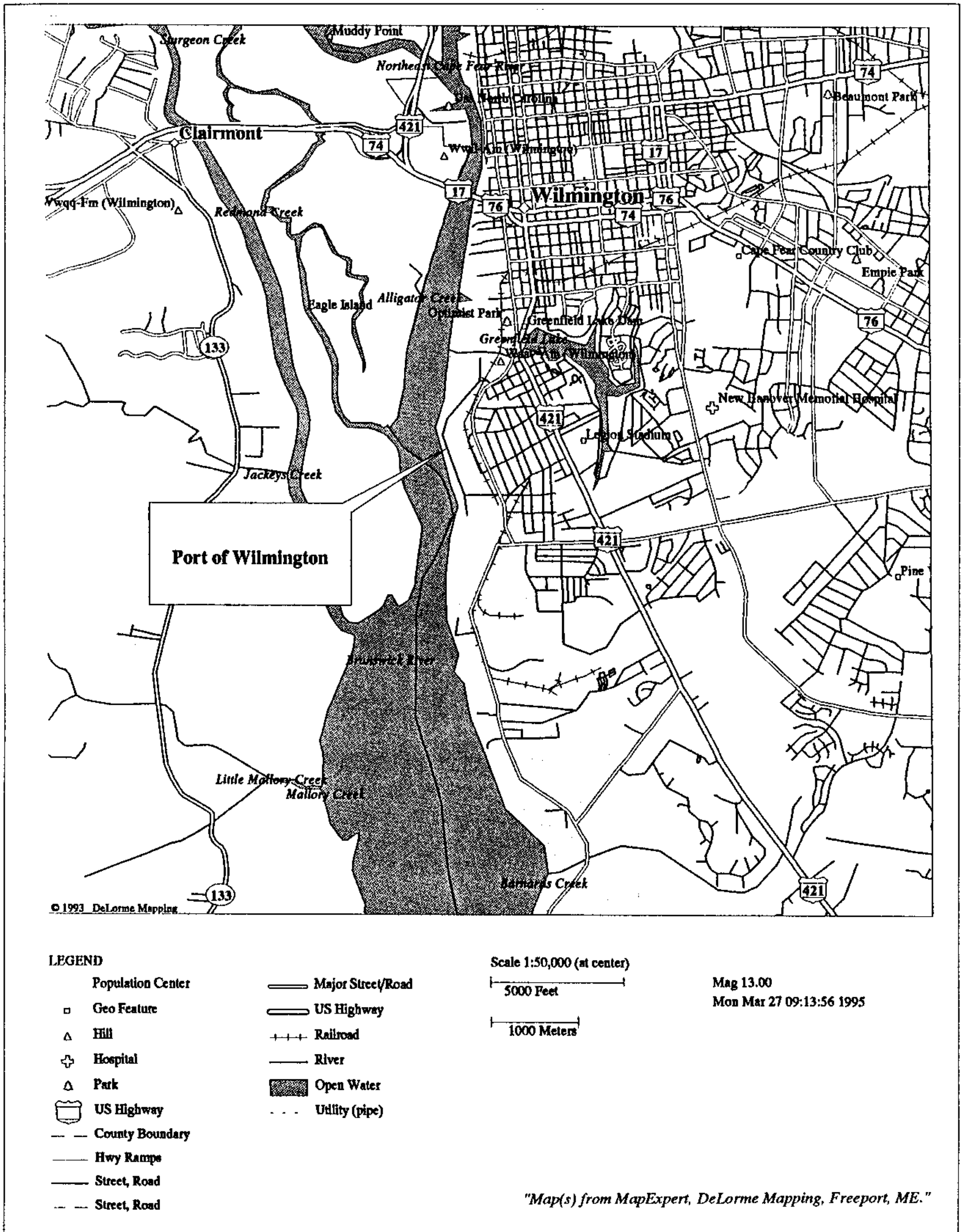


Figure 3-39 Port of Wilmington, NC

Environmental Conditions: There are no known environmentally sensitive areas in the immediate vicinity of the terminal, but due to resorts and recreational activity, there is heightened environmental awareness.

North Carolina has given the lower portion of the Cape Fear River three different stream classifications. From the Northeast Cape Fear River to the confluence with the Cape Fear River the waters are classified as SC-swamp. From the mouth of the Northeast Cape Fear to a point between Snow and Federal Points, the waters are classified as SC, and from Snow and Federal Points oceanward the waters are classified as SA. SC waters are tidal waters suitable for fishing, fish and wildlife propagation, secondary recreation, and other water uses requiring lower quality. The term "swamp" denotes waters with slow velocity. Class SA waters are suitable for shellfishing and primary recreation, as well as all of the activities approved for Class SC waters. According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Beaufort, NC, the Port of Wilmington is located in a low salinity estuarine habitat (generally 0.5 to 5 ppt) and tidal freshwater habitat. Below Wilmington at Campbell Island, the river changes to a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Cape Fear River near MOTSU changes once again to a high-salinity estuarine habitat (generally 16.5 to 30 ppt).

The lower Cape Fear River supports a large number of aquatic and terrestrial species. There are both invertebrate and fish species of commercial and recreational value found in the Cape Fear River near the Port of Wilmington. Species sought by commercial and recreational fishermen include flounder, trout, spot, croaker, bluefish, Spanish mackerel, and king mackerel. Shellfish sought include penaeid shrimp and blue crabs.

The Natural Heritage Program of the North Carolina Department of Environment, Health and Natural Resources reports that the area around the port has not been systematically inventoried for rare species. However, they also report that the lower Cape Fear River, from Wilmington to the mouth of the river at Smith Island, is brackish and contains numerous rare animals. The shortnose sturgeon (State and Federal Endangered Species) rarely occurs in the river, whereas manatees (State and Federal Endangered Species) occasionally occur, especially in the summer. American alligators (a designated threatened species) can be found in tributary streams. The freckled blenny, spinycheek sleeper, opossum pipefish, and marked goby are other rare marine fishes that inhabit the river.

There are many animals with special status in this area including various types of whales, sea turtles, and birds. State or Federally protected, endangered, or threatened aquatic species in this area include the shortnose sturgeon (fish), finback whale, Florida manatee, humpback whale, right whale, sei whale, and sperm whale (mammals), Arctic peregrine falcon, bald eagle, piping plover, red-cockaded woodpecker, wood stork (birds), and the American alligator, green sea turtle, hawksbill sea turtle, Kemp's ridley sea turtle, leatherback sea turtle, and the loggerhead sea turtle (reptiles and amphibians).

Climatic Conditions: The Port of Wilmington is located on the Cape Fear River, 42 km (26 mi) from the open Atlantic Ocean. This general area also includes MOTSU, which is also located on the Cape Fear River, north of Southport, NC, and south of Wilmington, NC. The elevation of this region is approximately 12 m (40 ft) above sea level, and is more variable than the coastal plain surrounding the Norfolk, VA area. The National Weather Service has been archiving meteorological information for this area since 1871.

The port is subject to hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code. For the Port of Wilmington, the Uniform Building Code provides a basic wind speed of about 160 km per hour (100 mph) (UBC, 1991). The port is located in a low seismic zone with an acceleration of 0.075 g.

The maritime location of the Wilmington area makes the climate unusually mild for its northern latitude. All wind directions from the east-northeast through the southwest have some moderating effect on the local climate, due to the relatively warm and cool ocean in the winter and summer seasons, respectively. The area rarely experiences cold episodes where the temperature falls below -18°C (0°F). However, cold air outbreaks do occur, causing sharp fluctuations in winter temperatures. Rainfall in the area is generally considered ample and evenly distributed throughout the year, with the bulk of the precipitation occurring during the summer months. The bulk of this rainfall is generally associated with afternoon and evening thunderstorms. In contrast, the winter rains tend to be of the slow, steady type, generally lasting one to two days. As is common at Atlantic coastal localities at this latitude, the late summer and early fall months bring the possibility of hurricanes and tropical storms to the Wilmington area. These storms are capable of generating high winds, above normal tides and torrential rains. The latter two are also capable of creating widespread local flooding of low-lying coastal areas (NOAA, 1992g).

Ethnic and Income Characteristics: Figure 3-40 shows the ethnic composition for the area surrounding the port at Wilmington. This figure shows the population residing within 16 km (10 mi) of the port according to 1990 data published by the U.S. Bureau of the Census. At the time of the 1990 census, African Americans made up about 33 percent of the total population, and approximately 93 percent of the minority population for the area surrounding the port. Figure 3-41 shows analogous information for low-income households residing within 16 km (10 mi) of the port. As discussed in Appendix A, the percentage of low-income households near the port is nearly the same as that for counties surrounding the port.

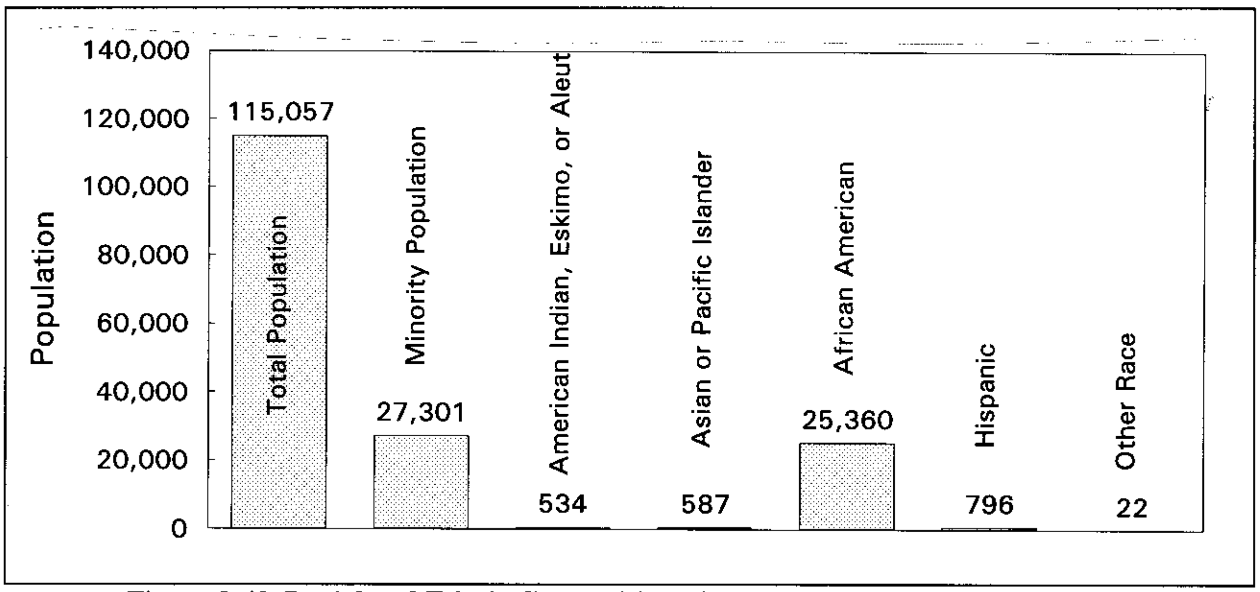


Figure 3-40 Racial and Ethnic Composition of the Minority Population Residing within 16 km (10 mi) of the Port of Wilmington

3.3 Management Site(s) Environments

This section describes the affected environment of the five potential DOE management sites for the foreign research reactor spent nuclear fuel. The five management sites are the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

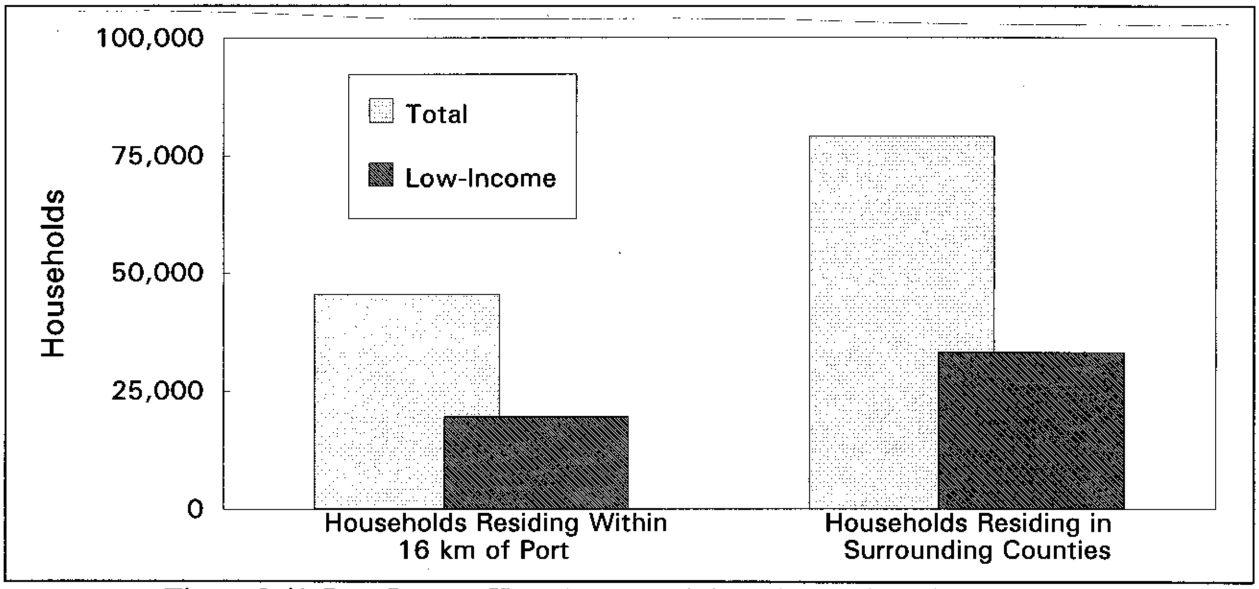


Figure 3-41 Low-Income Households Residing within 16 km (10 mi) of the Port of Wilmington

3.3.1 Description of the Affected Environment at the Savannah River Site

The Savannah River Site is a key DOE facility for research and production of special nuclear materials. The site was built in the early 1950's to produce the basic materials used in the fabrication of nuclear weapons. The DOE Savannah River Operations Office manages the Savannah River Site, and the Westinghouse Savannah River Company operates the site under contract to DOE. This section describes the potentially affected environment of the Savannah River Site. The location of the site is shown in Figure 3-42.

3.3.1.1 Geology

The Savannah River Site is located in the Upper Atlantic Coastal Plain physiographic province of western South Carolina, approximately 32 km (20 mi) southeast of the Fall Line, which separates the Piedmont and Coastal Plain provinces (Figure 3-42). The Coastal Plain in South Carolina is subdivided to include the Aiken Plateau, the Congaree Sand Hills, and the Coastal Terraces. The Coastal Plain consists of 213 to 366 m (700 to 1,200 ft) of gently seaward (southeast) dipping sands, clays, and limestones of Cretaceous and Tertiary age. These sediments are underlain by sandstones of Triassic age and older dense metamorphic and igneous basement rocks (Arnett et al., 1993). Coastal Plain sediments form a wedge of seaward-dipping and thickening unconsolidated and semi-consolidated sediments that begin at zero at the Fall Line and increase to more than 1,212 m (4000 ft) at the Continental Shelf. The Coastal Plain sediments underlying the Savannah River Site consist of sandy clays and clayey sands, with occasional beds of clean sand, gravel, clay, or carbonate. Two bioclastic limestone zones ranging from 0.6 m (2 ft) to 24 m (80 ft) occur within the Tertiary sequence. Most of the clastic sediments are unconsolidated, but thin semi-consolidated beds also occur (DOE, 1991a). The Triassic formations and older igneous and metamorphic rocks are hydrologically separated from the overlying Coastal Plain sediments by a regional aquitard (Arnett et al., 1993) (Figure 3-43).

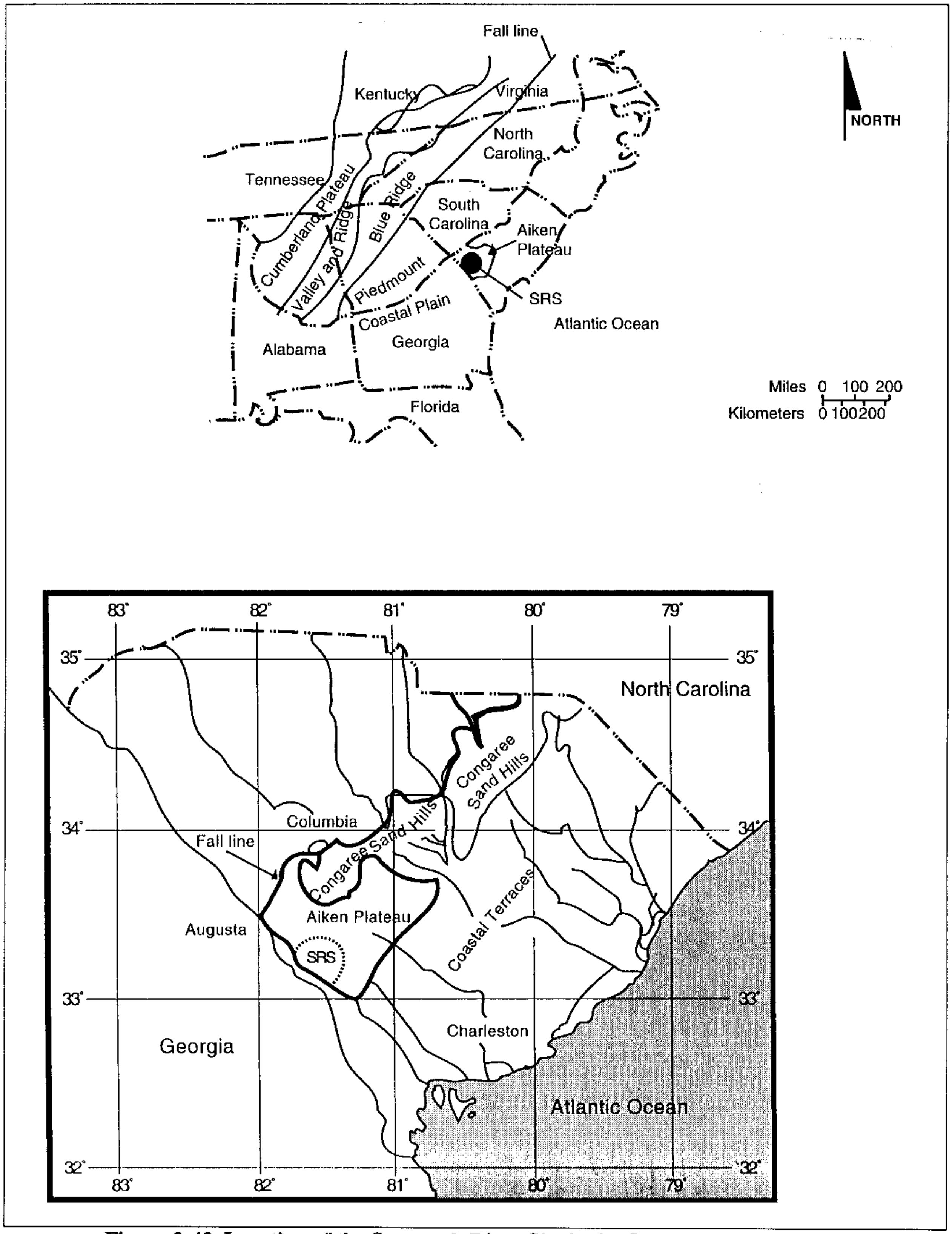


Figure 3-42 Location of the Savannah River Site in the Southern United States

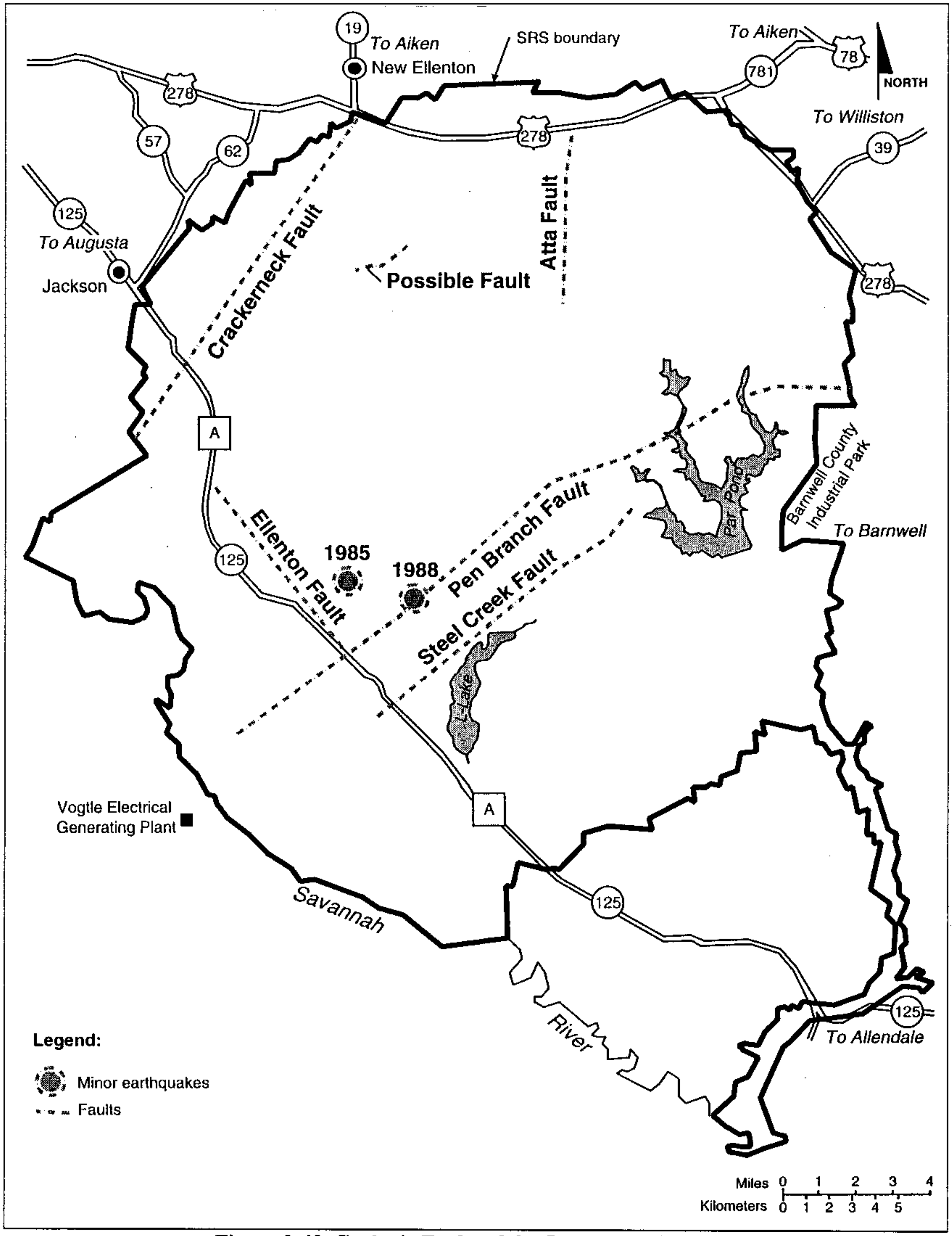


Figure 3-43 Geologic Faults of the Savannah River Site

3.3.1.2 Seismology and Volcanology

Seismicity in the Coastal Plain of South Carolina occurs in three distinct seismic zones near the Charleston area: Middleton Place Summerville, about 19 km (12 mi) northwest of Charleston; Bowman, about 59 km (37 mi) northwest of the Middleton Place-Summerville; and Adams Run, about 30 km (19 mi) southwest of the Middleton Place-Summerville (WSRC, 1993a). Of the three seismic zones within the Coastal Plain province, the Charleston area has been and remains the most seismically active. The Charleston area is also the most significant source of seismicity affecting the Savannah River Site, both in terms of maximum historic site intensity and the number of earthquakes felt in the area (WSRC, 1993a).

The closest offsite fault system is the Augusta Fault Zone, approximately 40 km (25 mi) from the Savannah River Site. In this fault zone, the Belair Fault has experienced the most recent movement, but is not considered capable of generating major earthquakes (DOE, 1987). There is no conclusive evidence of recent displacement along any fault within 320 km (200 mi) of the Savannah River Site, with the possible exception of the buried faults in the epicentral area of the 1886 Charleston, SC earthquake, approximately 144 km (90 mi) away (DOE, 1991a).

Two notable earthquakes have occurred within 320 km (200 mi) of the Savannah River Site. The first was a major earthquake in 1886 centered in the Charleston area, which had an estimated Richter magnitude of 6.8. The second earthquake was the Union County, SC earthquake of 1913, which had an estimated Richter magnitude of 6.0, and occurred about 160 km (100 mi) from the Savannah River Site (WSRC, 1993a).

Two earthquakes have occurred at the Savannah River Site during recent years. In June 1985, onsite instruments recorded an earthquake with a magnitude of 2.6 and a focal depth of about 1.0 km (0.6 mi) (DOE, 1995c). The epicenter was just west of the C- and K-areas. In August 1988, an earthquake of magnitude 2.0 and a focal depth of approximately 2.7 km (1.7 mi) occurred (Stephenson, 1988).

3.3.1.3 Hydrology

3.3.1.3.1 Surface Water

The Savannah River bounds the Savannah River Site on its southwestern border for about 32 km (20 mi), approximately 260 river km (160 river mi) from the Atlantic Ocean. At the Savannah River Site, the Savannah River flow averages about 283 m^3 per sec (74,760 gal per sec). Five principal tributaries to the Savannah River are found on the Savannah River Site: Upper Three Runs Creek, Fourmile Branch, Pen Branch, Steel Creek, and Lower Three Runs Creek (Figure 3-44). These tributaries drain almost all of the Savannah River Site. Each of these streams originates on the Aiken Plateau in the Coastal Plain, and descends 15 to 60 m (50 to 200 ft) before discharging into the river. The streams, which historically have received varying amounts of discharge from the Savannah River Site operations, are not commercial sources of water. The natural flow of the Savannah River Site streams ranges from less than 1 m^3 per sec (264 gal per sec) in smaller streams such as Pen Branch to 6.8 m^3 per sec (1,795 gal per sec) in Upper Three Runs. Three large upstream reservoirs - Hartwell, Richard B. Russell, and Strom Thurmond - minimize the effects of droughts and the impacts of low flow on downstream water quality and fish and wildlife resources in the Savannah River.

Surface Water Quality: The Savannah River, which forms the boundary between the States of Georgia and South Carolina, supplies potable water to several users. Upstream of the Savannah River Site, the river supplies domestic and industrial water needs for Augusta, GA, and North Augusta, SC. Downstream

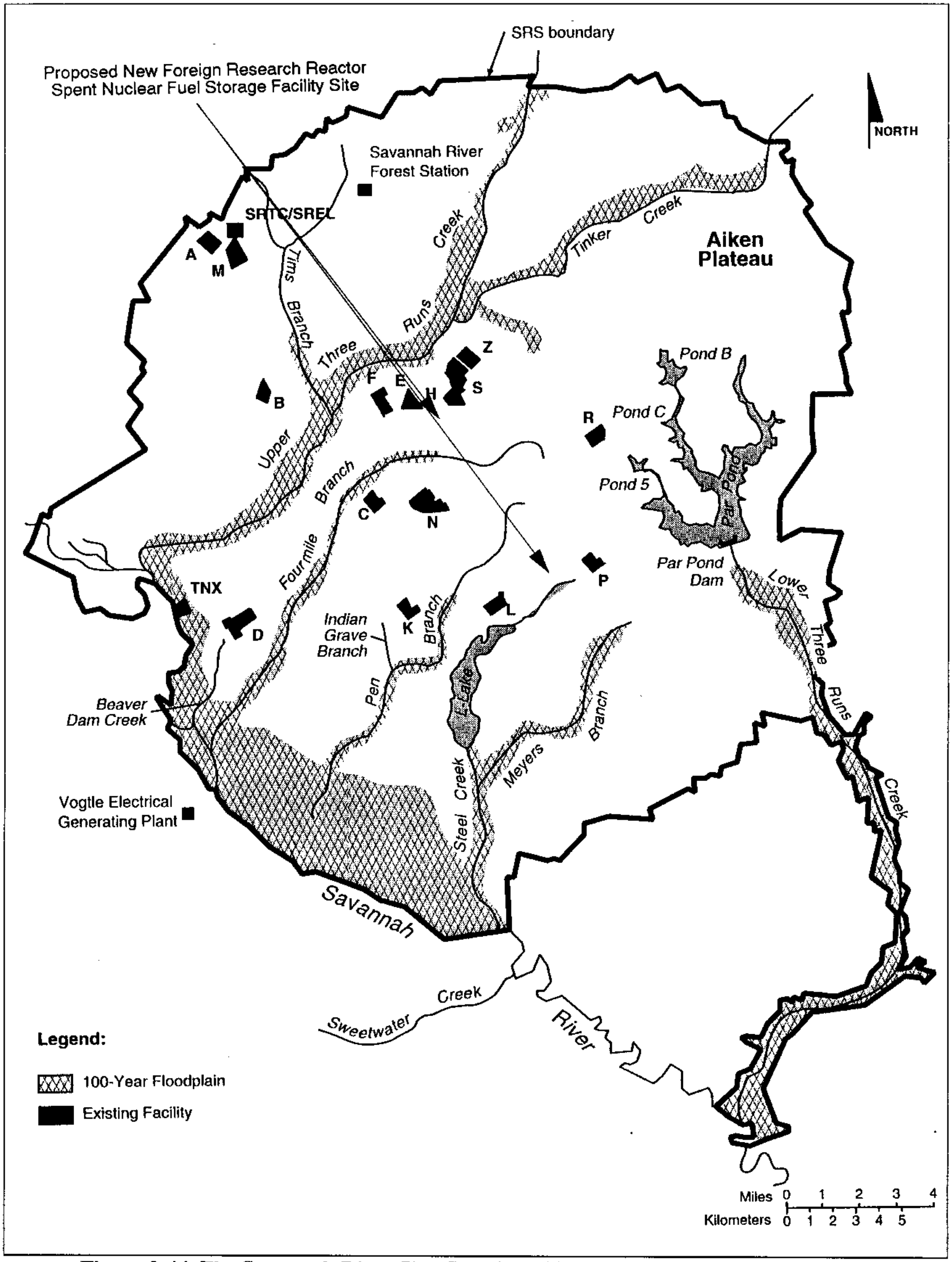


Figure 3-44 The Savannah River Site, Showing 100-Year Floodplain, Major Stream Systems and Facilities

of the Savannah River Site, the river supplies domestic and industrial water needs for Savannah, GA, and Beaufort and Jasper Counties in South Carolina. The South Carolina Department of Health and Environmental Control regulates the physical properties and concentrations of chemicals and metals in the Savannah River Site effluent under the National Pollutant Discharge Elimination System. This department also regulates chemical and biological water quality standards for the Savannah River Site waters. On April 24, 1992, the department changed the classification of the Savannah River and the Savannah River Site streams from "Class B waters" to "Freshwaters." The definitions of "Class B" waters and "Freshwaters" are the same, but the Freshwaters classification imposes a more stringent set of water quality standards (Arnett et al., 1993). Tables 4-10 and 4-11 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS list the characteristics of the Savannah River Site surface water quality (DOE, 1995c).

3.3.1.3.2 Groundwater

There are two hydrogeologic provinces in the subsurface beneath the Savannah River Site. The deepest Piedmont hydrogeologic province includes Paleozoic metamorphic and igneous basement rocks, and Triassic-aged lithified mudstone and sandstone. The Southeastern Coastal Plain hydrogeologic province lies above the Piedmont province and consists of a seaward thickening wedge of unconsolidated sediments of Late Cretaceous and Tertiary age. The Southeastern Coastal Plain hydrogeologic province is more important from a resource point of view because it holds an abundant supply of high-quality groundwater.

The sediments that make up the Southeastern Coastal Plain hydrogeologic province in west-central South Carolina are grouped into three major aquifer systems divided by two major confining systems, all of which are underlain by the Appleton confining system. The Appleton system separates the Southeastern Plain hydrogeologic province from the underlying Piedmont hydrogeologic province. Locally, individual aquifer and confining units are delineated. The complexly interbedded strata that form the three aquifer systems primarily consist of fine-to-coarse-grained sand and local gravel and limestone deposited under relatively high energy conditions in fluvial to shallow marine environments.

The water table receives water through rainfall percolating through the vadose zone. The deeper semi-confined aquifers receive water from groundwater flow into the Savannah River Site from offsite or from water flowing from aquifers above or below. The direction of groundwater flow in the vadose zone is predominantly downward, but some lateral flow occurs because of clay lenses in the soil. The flow of groundwater in the water table and deeper semi-confined aquifers is controlled by the hydraulic properties of the sediments (e.g., conductivity) and the proximity to streams. Savannah River ultimately receives all groundwater that flows beneath the Savannah River Site, and no contaminated groundwater is flowing off of the Savannah River Site.

Groundwater Quality: The quality of groundwater in the principal hydrologic systems beneath the Savannah River Site depends on both the source of the water and the inorganic and biochemical reactions that take place along its flowpath. Quality is strongly influenced by the chemical composition and mineralogy of the enclosing geologic materials (WSRC, 1993b). In general, the quality of the groundwater in the Coastal Plain sediments at the Savannah River Site and the surrounding areas is suitable for most domestic and industrial purposes. The waters are dilute with respect to total dissolved solids concentrations, which range from less than 10 mg per L to about 150 to 200 mg per L. The pH values range from as low as 4.9 to a maximum value of 7.7 (where the groundwater is in contact with limestone). Due to the low solids content of the waters and the frequently low pH values, many of the waters are corrosive to metal surfaces. High dissolved iron concentrations can also be of concern in some units. An onsite degasification and filtration process raises the pH and removes iron in domestic water supplies where necessary (WSRC, 1993b). Table 4-12 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS summarizes the Savannah River Site groundwater quality data, and Table 4-13 lists data for radiological constituents (DOE, 1995c).

The groundwater beneath 5 to 10 percent of the Savannah River Site has been contaminated by industrial solvents, metals, tritium, or other constituents used or generated on the site. Figure 3-45 shows the locations of facilities monitored by the Savannah River Site and areas with constituents that exceeded drinking water standards in 1992. In general, contaminated groundwater at the Savannah River Site is beneath a few facilities, and the contaminants reflect operations and chemical processes at those facilities. For example, contaminants in the groundwater beneath A- and M-Areas include chlorinated volatile organics, radionuclides, metals, and nitrate. At F- and H-Areas, contaminants in the groundwater include tritium and other radionuclides, metals, nitrate, chlorinated volatile organics (at values much smaller than found at A- and M-Areas), and sulfate. The groundwater beneath the Sanitary Landfill contains chlorinated volatile organics, radionuclides, and metals. The groundwater beneath all the reactor areas except R-Area contains tritium, other nuclides, metals, and chlorinated volatile organics, and at R-Area, groundwater contaminants include radionuclides and cadmium. The groundwater beneath D-Area contains metals, radionuclides, sulfate, and chlorinated volatile organics. At TNX-Area, the groundwater contains chlorinated volatile organics, radionuclides, and nitrate (Arnett et al., 1993).

The McQueen Branch aquifer, which becomes shallower toward the Fall Line, forms the base for most municipal and industrial water supplies in Aiken County. Toward the coast, in Allendale and Barnwell Counties, this aquifer exists at increasingly greater depths. As a consequence, the shallower Gordan aquifer supplies some municipal, industrial, and agricultural users. The Gordan and Upper Three Runs Creek aquifers are the primary sources of domestic water supplies in the vicinity of the Savannah River Site for rural non-municipal water. DOE has identified 56 major municipal, industrial, and agricultural groundwater users within 32 km (20 mi) of the center of the Savannah River Site (DOE, 1987). The total pumpage for these users is about 136,260 m^3 per day (36 million gal per day).

Excellent quality groundwater is abundant in this region of South Carolina from many local aquifer units. As a result, the South Carolina Department of Health and Environmental Control has classified all aquifers in the State as Class GB (DOE, 1995c), or U.S. Environmental Protection Agency Class II, meaning that the aquifers can provide resource-quality water, but are not the sole source of supply (as are South Carolina Class GA or U.S. Environmental Protection Agency Class 1 aquifers) (DOE, 1991a).

3.3.1.4 Meteorology

Wind: Figure 3-46 shows annual wind direction frequencies and wind speeds for the Savannah River Site from 1987 through 1991. The maximum wind directional frequencies are from the northeast and west-southwest. The average wind speed for this 5-yr period was 3.8 m per sec (8.5 mph). Calm winds (less than 2 m per sec or 4.5 mph) occurred less than 10 percent of the time during the 5-yr period. Seasonally, wind speeds were greatest during the winter at 4.1 m per sec (9.2 mph), and lowest during the summer at 3.4 m per sec (7.6 mph) (Shedrow, 1993). Winter snow storms in the Savannah River Site area occasionally bring strong and gusty surface winds with speeds as high as 32 m per sec (72 mph). Thunderstorms can generate winds with speeds as high as 18 m per sec (40 mph) and even stronger gusts. The fastest wind speed recorded at Augusta between 1950 and 1986 was 37 m per sec (83 mph) (DOE, 1995c).

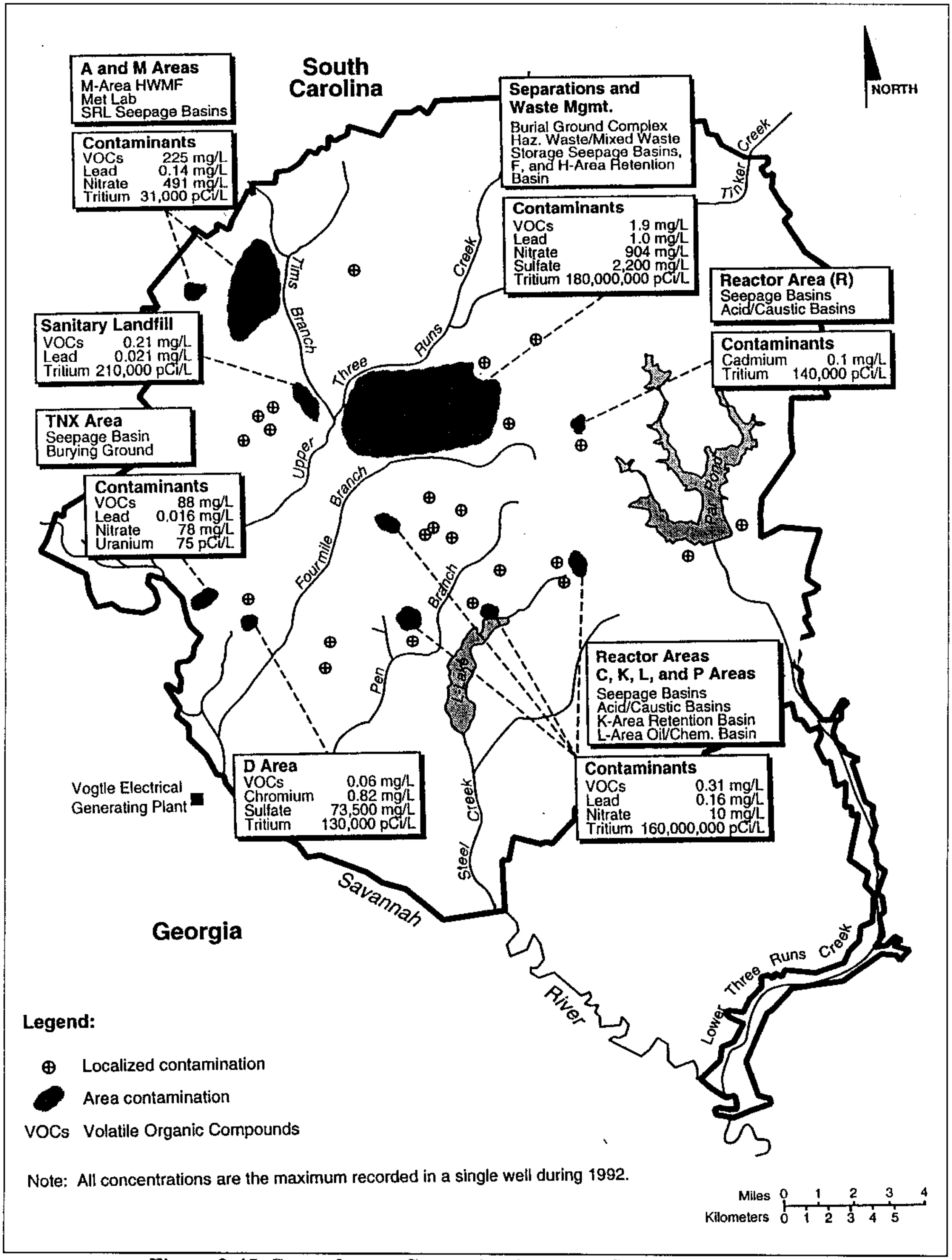
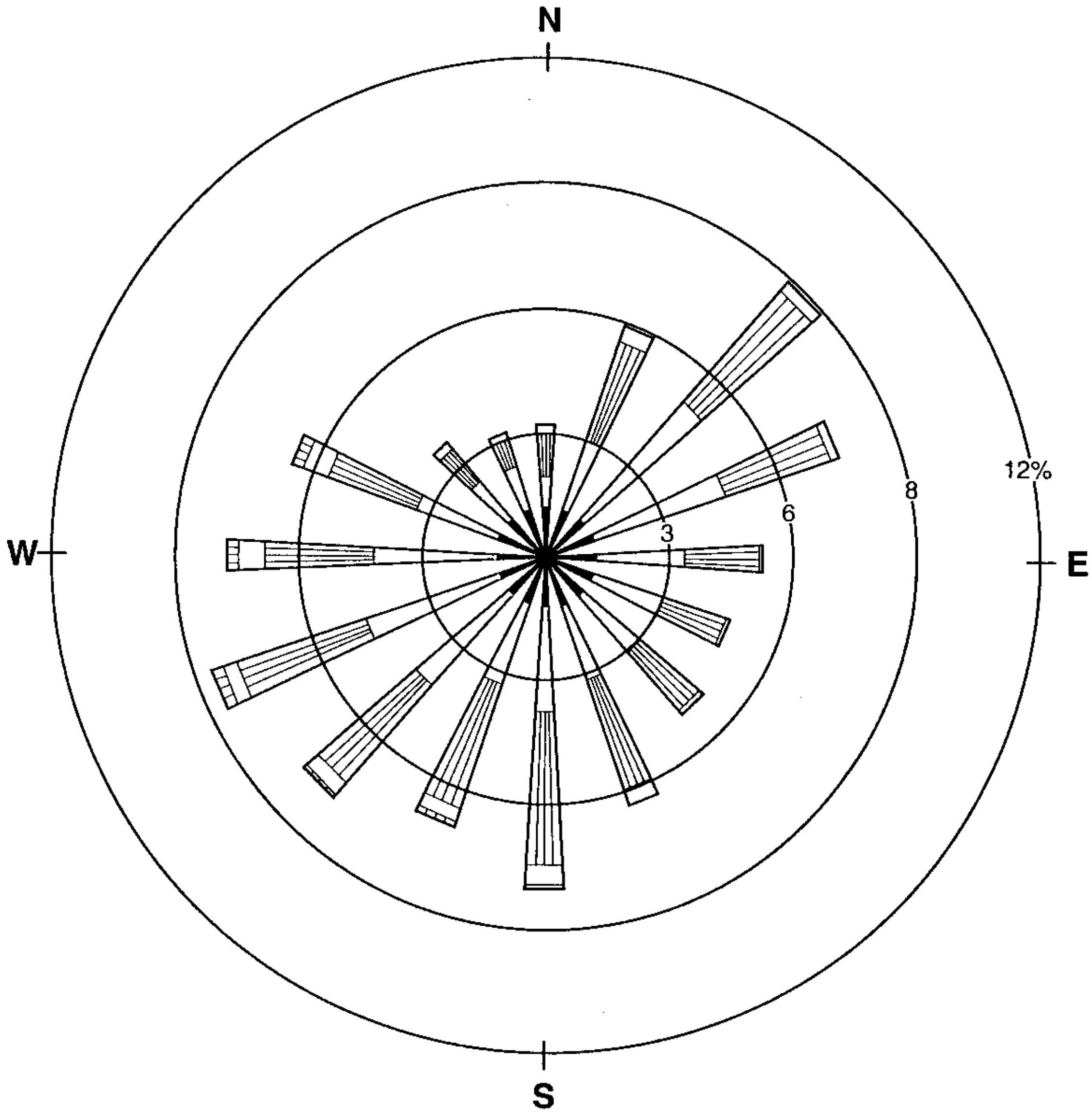
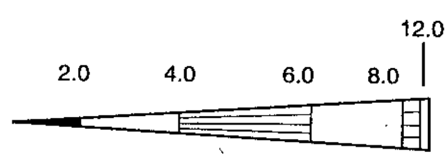


Figure 3-45 Groundwater Contamination at the Savannah River Site



The wind rose plot shows percent occurrence frequencies of wind direction and speed at SRS. It is based on a composite of hourly averaged wind data from the SRS meteorological tower network for the 5-year period 1987-1991. Measurements were taken from 200 feet above ground. Directions indicated are *from* which the wind blows.



Wind Speed Class Boundaries (meters/second)

Figure 3-46 Wind Rose for the Savannah River Site (1987-1991)

Temperature and Humidity: The annual average temperature at the Savannah River Site is 17.8°C (64°F), and monthly averages range from a low of 7.22°C (45°F) in January to a high of 27.2°C (81°F) in July. Average daily relative humidity ranges from a maximum of 90 percent in the morning to a minimum of 43 percent in the afternoon on an annual basis.

Precipitation: The average annual precipitation at the Savannah River Site is approximately 121.9 cm (48 in). Precipitation distribution is fairly even throughout the year, with the highest precipitation in the summer [36.1 cm (14.2 in)] and the lowest in autumn [22.5 cm (8.8 in)] (Arnett et al., 1993). Snowfall has occurred in the months of October through March, with the average annual snowfall at 3.0 cm (1.2 in). Large snowfalls are rare (DOE, 1995c).

The area encompassing the Savannah River Site experiences an average of 56 thunderstorm days per year. From 1954 to 1983, 37 tornadoes were reported for a one-degree square of latitude and longitude that includes the Savannah River Site. This frequency of occurrence is equivalent to an average of about one tornado per year. The estimated probability of a tornado striking a point on the Savannah River Site is 0.00007 per year, which is less than one in ten thousand (DOE, 1995c). Since operations began at the Savannah River Site in 1953, nine tornadoes have been confirmed on or near the site. Winds exceeding hurricane force have been observed only once at the Savannah River Site (Hurricane Gracie in 1959) (Shedrow, 1993).

Atmospheric Dispersion: Based on measurements at onsite meteorological stations, dispersion conditions in the Savannah River Site region were classified unstable approximately 56 percent of the time, neutral 23 percent of the time, and stable about 21 percent of the time. On an annual basis, inversion conditions occur 21 percent of the time at the Savannah River Site (Shedrow, 1993).

Air Quality: The local air quality management region which includes the Savannah River Site is in attainment with National Ambient Air Quality Standards for criteria pollutants, which include sulfur dioxide, nitrogen oxides, particulate matter, lead, ozone (as volatile organic compounds), and carbon monoxide (EPA, 1993a). This region has a Class II designation under Prevention of Significant Deterioration regulations (EPA, 1993b), which allows moderate industrial growth to occur. No areas within an 80 km (50 mi) radius of the site are designated as Prevention of Significant Deterioration Class I (e.g., national parks, wildlife refuge). Class I areas place severe restrictions on new sources that might affect ambient air quality. The States of South Carolina and Georgia perform ambient air monitoring near the Savannah River Site, and have reported no significant exceedances of National Ambient Air Quality Standards.

In the Savannah River Site region, airborne radionuclides originate from natural resources (terrestrial or cosmic), worldwide fallout, and the Savannah River Site operations. The Savannah River Site maintains a network of air monitoring stations on and around the site to determine the concentrations of radioactive particulates and aerosols in the air (Arnett et al., 1993). Table 4-6 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS lists average and maximum atmospheric radionuclide particulate concentrations at the Savannah River Site boundary, and background [160 km (100 mi)] monitoring locations in 1991 (DOE, 1995c). Tritium is the only radionuclide of the Savannah River Site origin that can be detected routinely in offsite air samples above background.

3.3.1.5 Ecology

When the U.S. Government acquired the Savannah River Site in 1951, the site was approximately two-thirds forested and one-third cropland and pasture (Dukes, 1984). At present, more than 90 percent of the Savannah River Site is forested. With the exception of the Savannah River Site production and support areas, natural succession has reclaimed other previously disturbed areas. Satellite imagery of the site shows a circle of wooded habitat within a matrix of cleared uplands and narrow forested riparian corridors. The Savannah River Site provides nearly 73,250 ha (181,000 acres) of contiguous forested cover broken only by unpaved secondary roads, transmission line corridors, and a few paved primary roads. Carolina bays, the Savannah River swamp, and several relatively intact longleaf pine-wiregrass communities make important contributions to the biodiversity of the region.

The Savannah River Site is near the transition area between the oak-hickory-pine forest and the southern mixed forest. As a consequence, species typical of both associations occur (Dukes, 1984). A variety of vascular plant communities occur in the upland areas. Typically, scrub oak communities occur on the drier, sandier areas. Longleaf pine, turkey oak, bluejack oak, blackjack oak, and dwarf post oak dominate these communities, which typically have understories of wire grass and huckleberry. Oak-hickory hardwood communities occur on more fertile, dry uplands, and characteristic species are white oak, post oak, southern red oak, mockernut hickory, pignut hickory, and loblolly pine, with an understory of sparkleberry, holly, greenbriar, and poison ivy (DOE, 1995c).

The Savannah River Site has provided excellent habitat to wildlife associated with the wetlands of the Savannah River and the pine-dominated sandhills of coastal South Carolina. Furbearers such as gray fox, raccoon, opossum and beaver are relatively common throughout the Savannah River Site. Game species such as gray and fox squirrel, cottontail rabbit, and wild turkey are also common. The Savannah River Site contains suitable habitat for white-tailed deer and feral hogs, as well as other faunal species common to the mixed pine/hardwood forests of South Carolina.

The Savannah River Site has extensive, widely distributed wetlands, most of which are associated with floodplains, creeks, and impoundments. The southwestern Savannah River Site boundary adjoins the Savannah River for approximately 32 km (20 mi). The river floodplain supports an extensive swamp, covering about 4,916 ha (12,148 acres) of the site. At present, the swamp forest consists of second-growth bald cypress, black gum, and other hardwood species (USDA, 1991). Five major streams drain the Savannah River Site, and eventually flow into the Savannah River. Each stream has floodplains characterized by bottomland hardwood forests or scrub-shrub wetlands in varying stages of succession. Dominant species include the red maple, box elder, bald cypress, water tupelo, sweetgum, and black willow (DOE, 1995c). Carolina bays, unique wetland features of the southeastern United States, are islands of wetland habitat dispersed throughout the uplands of the Savannah River Site. The more than 200 bays on the site exhibit extremely variable hydrology and a range of plant communities from herbaceous marsh to forested wetland (Shields et al., 1982; Schalles et al., 1989).

Threatened, Endangered, and Candidate Plant and Animal Species: Threatened, endangered, and candidate plant and animal species on the Savannah River Site include 5 bird species, 1 mammal species, 5 amphibian species, 5 reptile species, 1 fish species, 2 invertebrate species, and 19 plant species. The following Federally listed endangered animals are known to occur on the Savannah River Site or in the Savannah River adjacent to the site: the red-cockaded woodpecker, the southern bald eagle, the wood stork, and the shortnose sturgeon (DOE, 1995c). Researchers have found one Federally listed endangered plant species, the smooth coneflower, on the Savannah River Site, along with several Federally listed Category 2 species, and several listed species (Knox and Sharitz, 1990).

3.3.1.6 Land Use

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in western South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, GA. The Savannah River Site, which is bordered by the Savannah River to the southwest, includes portions of Aiken, Barnwell, and Allendale

Counties. Land use on the Savannah River Site can be grouped into three major categories: forest/undeveloped, water/wetlands, and developed facilities. Ninety-six percent of the Savannah River Site area, about 73,450 ha (181,500 acres), is undeveloped (USDA, 1991). Approximately 90 percent of this area is forested (Cummins et al., 1990). In 1972, DOE designated the Savannah River Site as a National Environmental Research Park. At present, approximately 57 km^2 (22 mi^2), or 7 percent of the Savannah River Site area is designated as "Set-Asides," which are areas specifically protected for environmental research activities that are coordinated either through the University of Georgia Savannah River Ecological Laboratory or the Savannah River Technology Center (Cummins et al., 1990). At present, administrative production and support facilities occupy approximately 5 percent of the total the Savannah River Site land area.

Land bordering the Savannah River Site is primarily forest and agricultural. There is also a significant amount of open water and forested wetlands along the Savannah River Valley. Urbanized and industrial areas are the only other significant use of land in the vicinity (Figure 3-47). None of the three counties in which the Savannah River Site is located has zoned any of the site land. The only adjacent area with any zoning is the Town of New Ellenton, which has two zoning categories for lands that bound the Savannah River Site, urban development and residential development. The closest residences to the Savannah River Site boundary include several within 61 m (200 ft) of the site perimeter to the west, north, and northeast.

The Savannah River Site is a controlled area, with public access limited to through traffic on South Carolina Highway 125 (the Savannah River Site Road A), U.S. Highway 278, the Savannah River Site Road 1, and the CSX railway. The Savannah River Site does not contain any public recreation facilities. However, the Savannah River Site conducts controlled deer and feral hog hunts each fall, from mid-October through mid-December. The intent of the hunts is to control the resident populations of these animals and to reduce animal-vehicle accidents on the Savannah River Site roads.

3.3.1.7 Noise

The major noise sources at the Savannah River Site are found primarily in developed operational areas, and include various facilities, equipment, and machines (e.g., cooling towers, transformers, engines, pumps, boilers, steam vents, paging systems, construction and materials-handling equipment, and vehicles). Major noise sources outside the operational areas consist primarily of vehicles and railroad operations. Previous studies have analyzed noise impacts of existing the Savannah River Site operational activities (DOE, 1995c; DOE, 1991a; DOE, 1990a; DOE, 1993d). These studies concluded that, because of the remote locations of the Savannah River Site operational areas, there are no known conditions associated with existing onsite noise sources that adversely affect individuals at offsite locations. Some disturbance of wildlife activities might occur on the Savannah River Site as a result of hunting activities and construction activities. Noise limits are established for the workplace to protect workers' hearing in accordance with Occupational Health and Safety Administration standards. Existing Savannah River Site-related noise sources of importance to the public are those associated with the transportation of people and materials to and from the site. These sources include trucks, private vehicles, and freight trains. In addition, a portion of the air cargo and business travel using commercial air transport through the airports at Augusta, GA, and Columbia, SC, is attributable to the Savannah River Site operations. The States of Georgia and South Carolina, and the counties in which the Savannah River Site is located, have not established regulations that specify acceptable community noise levels, with the exception of a provision of the Aiken County Nuisance Ordinance, which limits daytime and nighttime noise by frequency band (Aiken County, 1991).

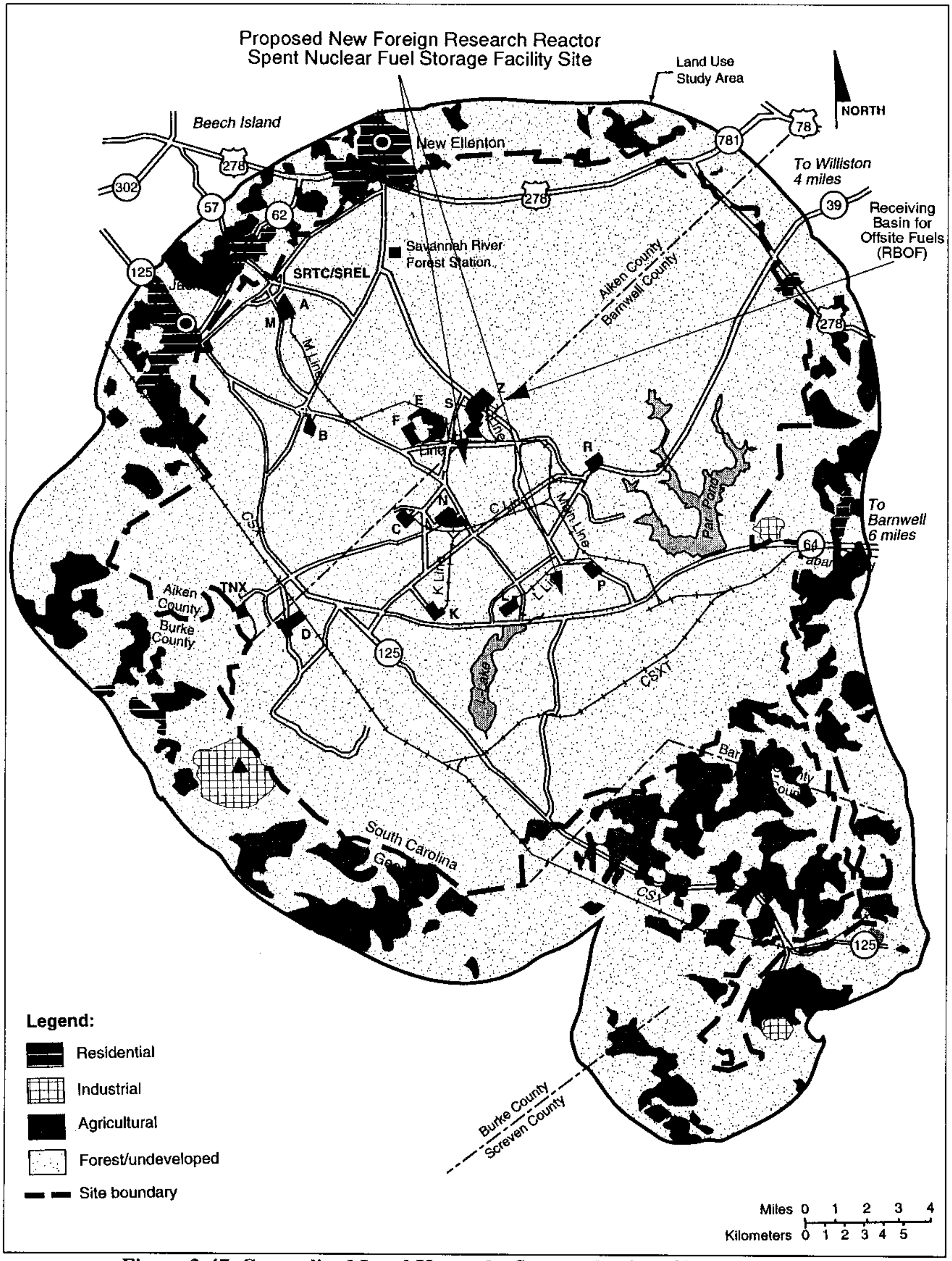


Figure 3-47 Generalized Land Use at the Savannah River Site and Vicinity

Noise from the Savannah River Site Traffic: During a normal week, about 20,000 employees travel to the Savannah River Site each day in private vehicles from surrounding communities. Both Government-owned and private trucks pick up and deliver materials at the site. The contribution of the Savannah River Site operations to traffic volumes along SC 125 and SC 19, especially during peak traffic periods, affects noise levels in the towns of New Ellenton and Jackson and the city of Aiken. Noise measurements taken during 1989 and 1990 along SC 125 in the town of Jackson (at a point about 15 m or 50 ft from the roadway) indicate that the one-hour equivalent sound level from traffic ranged from 48 to 72 decibels. The estimated day/night average sound level along this route was 66 decibels for summer and 69 decibels for winter. Similarly, noise measurements along SC 19 in the town of New Ellenton indicate that the one-hour equivalent sound level from traffic ranged from 53 to 71 decibels. The estimated day/night average sound level along this route was 66 decibels for both summer and winter (HNUS, 1990). Employment at the Savannah River Site has increased by about 17 percent since 1989, potentially causing increases in traffic noise, especially during peak traffic periods (approximately between 6:30 and 8:30 a.m., and between 3:30 and 5:30 p.m. corresponding to major shift changes). Since some residences and at least two schools are within 30 to 60 m (100 to 200 ft) of these routes, some annoyance to members of the public residing along these highways can occur based on the relationship between the day/night average sound level (Schultz 1978; FICON, 1992).

Noise from Railroad Traffic: Approximately 18 trains per day pass through the Savannah River Site on the CSX line, with five trains delivering shipments to the Savannah River Site. Noise sources from rail transport include diesel engines, wheel-track contact, and whistle-warnings at rail crossings.

3.3.1.8 Transportation

The Savannah River Site is surrounded by a system of Interstate highways, U.S. highways, State highways, and railroads. The regional transportation networks service the four South Carolina counties (Aiken, Allendale, Bamberg, and Barnwell), and two Georgia counties (Columbia and Richmond) that generate about 90 percent of the Savannah River Site commuter traffic (DOE, 1995c). Two major railroads—CSX Transportation and Norfolk Southern Corporation—also serve the Savannah River Site vicinity. Norfolk Southern serves Augusta and Savannah, GA, as well as Columbia and Charleston, SC. CSX serves the same locations and the Savannah River Site. Figure 3-48 shows the regional transportation infrastructure.

Two Interstate highways serve the Savannah River Site area. Interstate 20 (I-20) provides a primary east-west corridor and I-520 links I-20 with Augusta, GA. U.S. Highways 1 and 25 are principal north-south routes, and U.S. 78 provides east-west connections. Several other highways (U.S. 221, U.S. 301, U.S. 321, and U.S. 601) provide additional transport routes in the region. Several State routes provide direct access to the Savannah River Site. From the northwest and north, access is provided by SC 125 and SC 19, respectively, and SC 125 is open to through traffic. Access to the site is provided from the northeast by SC 39, from the east by SC 64, and from the southeast by SC 125. These are all two-lane roads. The public has access to U.S. 278 and SC 125, but only the Savannah River Site employees are permitted access to the site on the other routes.

The Savannah River Site transportation infrastructure consists of more than 230 km (143 mi) of primary roads, 1,931 km (1,200 mi) of unpaved secondary roads, and 103 km (64 mi) of railroad track (DOE, 1995c). These roads and railroads provide connections among the various Savannah River Site facilities and offsite transportation linkages. Figure 3-49 shows the Savannah River Site network of primary roadways and access points, and the Savannah River Site railway system.

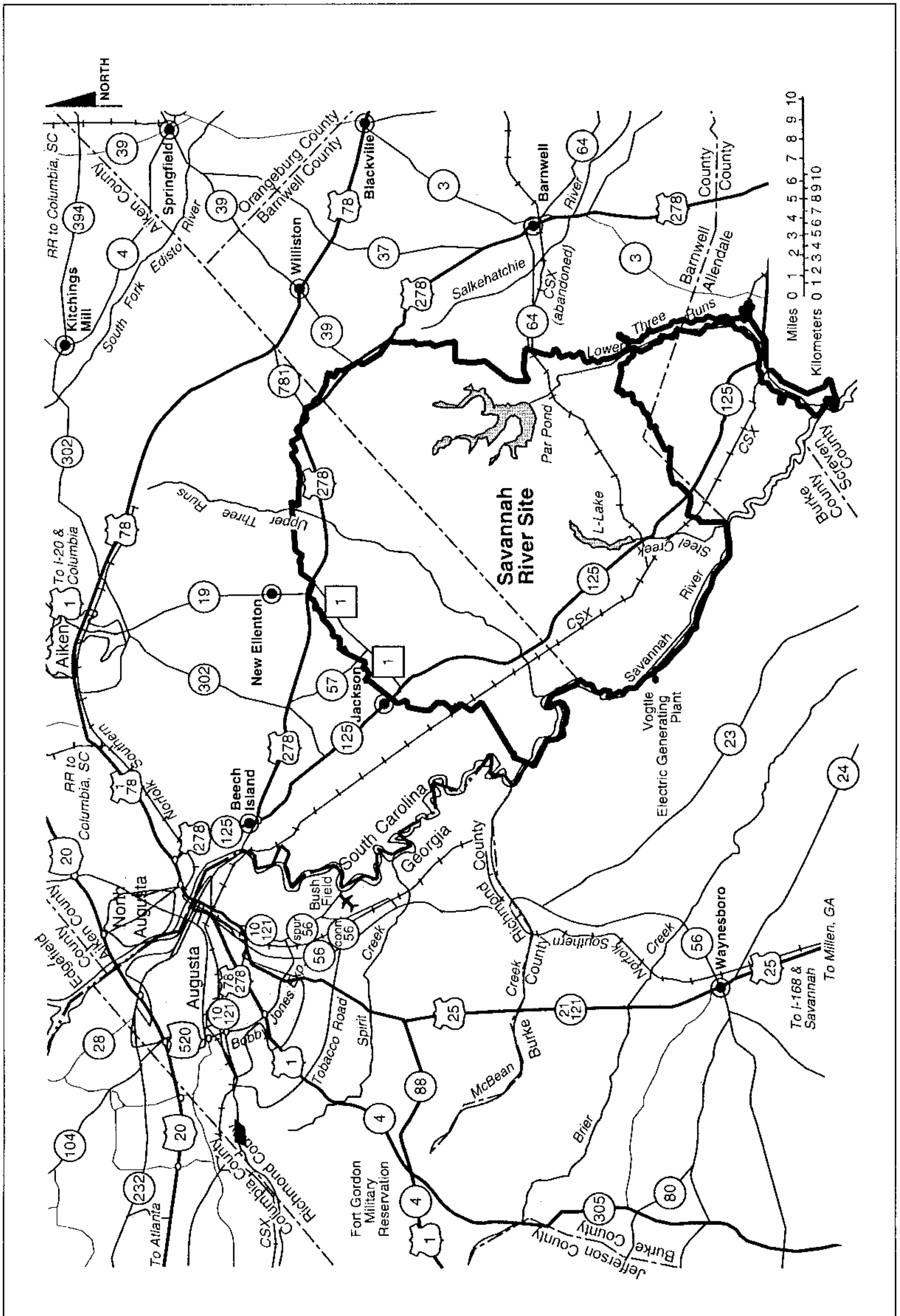


Figure 3-48 Regional Transportation Infrastructure

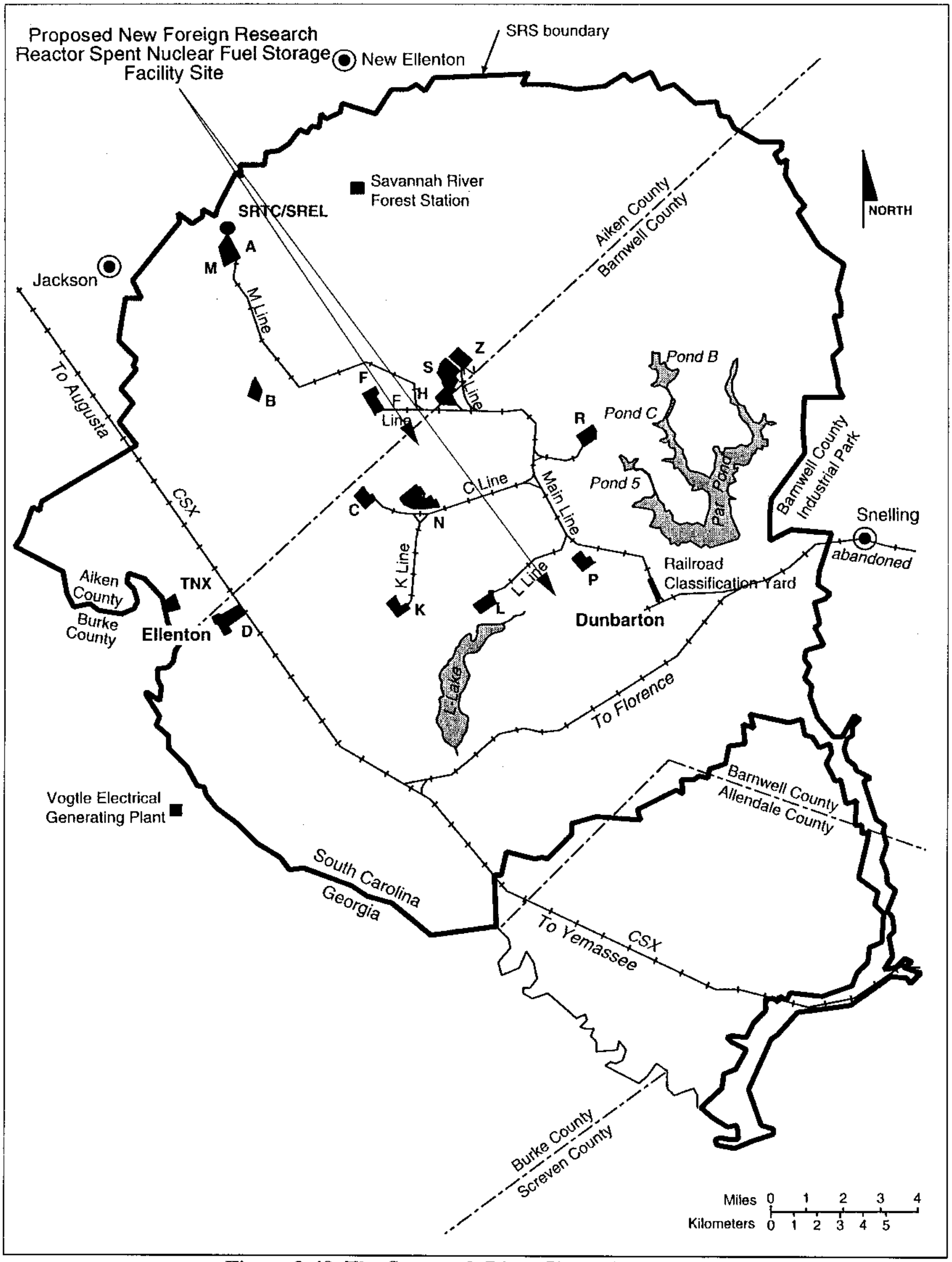


Figure 3-49 The Savannah River Site Railroad Lines

Two major public highways traverse the Savannah River Site: SC 125 and U.S. 278. SC 125 connects Allendale, SC, to Augusta, GA, by crossing the site in a northwest-to-southeast direction. U.S. 278 also connects Augusta and Allendale, but its route generally follows the northern and eastern Savannah River Site boundaries. In general, the primary Savannah River Site roadways are in good condition, and are smooth and free from potholes. Typically, wide, firm shoulders border roads that are either straight or have wide gradual turns. Intersections are well marked for both traffic and safety identification, and are sufficiently cleared of trees and brush that might obstruct a driver's view of oncoming traffic. Railings along the side of the roadways offer protection at appropriate locations from dropoffs or other hazards. In general, the roadways are lighted only at gate areas and near major facilities.

In general, heavy traffic occurs early in the morning and late in the afternoon when workers from surrounding communities commute to and from the Savannah River Site. During working hours, official vehicles and logging trucks constitute most of the traffic. At any time, as many as 60 logging trucks, which can impede traffic, might be operating on the Savannah River Site, with an annual average of about 25 trucks per day. Table 4-16 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS provides data on traffic counts for various roads and access points around the Savannah River Site (DOE, 1995c).

Railroads on the Savannah River Site include both CSX tracks and the Savannah River Site rolling stock and tracks. Two routes of the CSX distribution system run through the Savannah River Site: a line between Florence, SC, and Augusta, GA, and a line between Yemassee, SC, and Augusta. The two lines join on the site near the L-Lake dam (Figure 3-49). Early in 1989, CSX discontinued service on the line from the Savannah River Site junction to Florence. The 103 km (64 mi) of the Savannah River Site railroad tracks are well maintained. The rails and crosslines are in good condition, and the track lines are clear of vegetation and debris. Significant clear areas border the tracks on both sides. Intersections of railroads and roadways are marked by railroad crossing signs with lights where appropriate. The Savannah River Site rail classification yard is east of P-Reactor. This eight-track facility sorts and redirects railcars. Deliveries of the Savannah River Site shipments occur at two onsite rail stations at the former towns of Ellenton and Dunbarton. From these stations, a Savannah River Site engine moves the railcars to the appropriate receiving facility. The Ellenton station, which is on the main Augusta-Yemassee line, is the preferred delivery point. The Dunbarton station, which is on the discontinued portion of the Augusta-Florence line, receives less use.

3.3.1.9 Socioeconomics

The Savannah River Site region of influence includes Aiken, Allendale, Bamberg, and Barnwell Counties in South Carolina, and Columbia and Richmond Counties in Georgia. Between 1980 and 1990, total employment in the region of influence increased from 139,504 to 199,161, an average annual growth rate of approximately 5 percent. Table 4-1 of Appendix C, Volume 1 of the Programmatic SNF&INEL Final EIS lists projected employment data for the six-county region of influence. As shown, by the year 2000, employment levels should increase 27 percent to approximately 253,000. The unemployment rates for 1980 and 1990 were 7.3 percent and 4.7 percent, respectively (DOE, 1995c).

In 1990, employment at the Savannah River Site was 20,230, representing 10 percent of the region of influence employment (DOE, 1993d). In Fiscal Year 1992, employment at the Savannah River Site increased approximately 15 percent to 23,351, with an associated payroll of more than $1.1 billion. From 1980 to 1990, the labor force in the six-county region of influence grew 39 percent, from 150,551 to 208,984. In 1990, 75.3 percent of the region of influence labor force lived in Richmond and Aiken Counties, SC. Current projections call for the region's labor force to increase to approximately 257,000 workers by 1995 (DOE, 1995c).

Between 1980 and 1990, population in the region of influence increased 13 percent, from 376,058 to 425,607. More than 88 percent of the 1990 population lived in Aiken (28.4 percent), Columbia (15.5 percent), and Richmond (44.6 percent) Counties. According to 1990 census data, the estimated average number of persons per household in the six-county region was 2.72, and the median age of the population was 31.2 years (DOE, 1995c). Based on 1990 census population data, the general ethnic composition of the immediate area of influence, which is within an 80 km (50 mi) radius of the site, is shown in Figure 3-50. Low-income households are presented in Figure 3-51. Low-income households are those with incomes of 80 percent or less than the median income of the counties. As indicated in this figure, approximately 42 percent of the total households are low-income households.

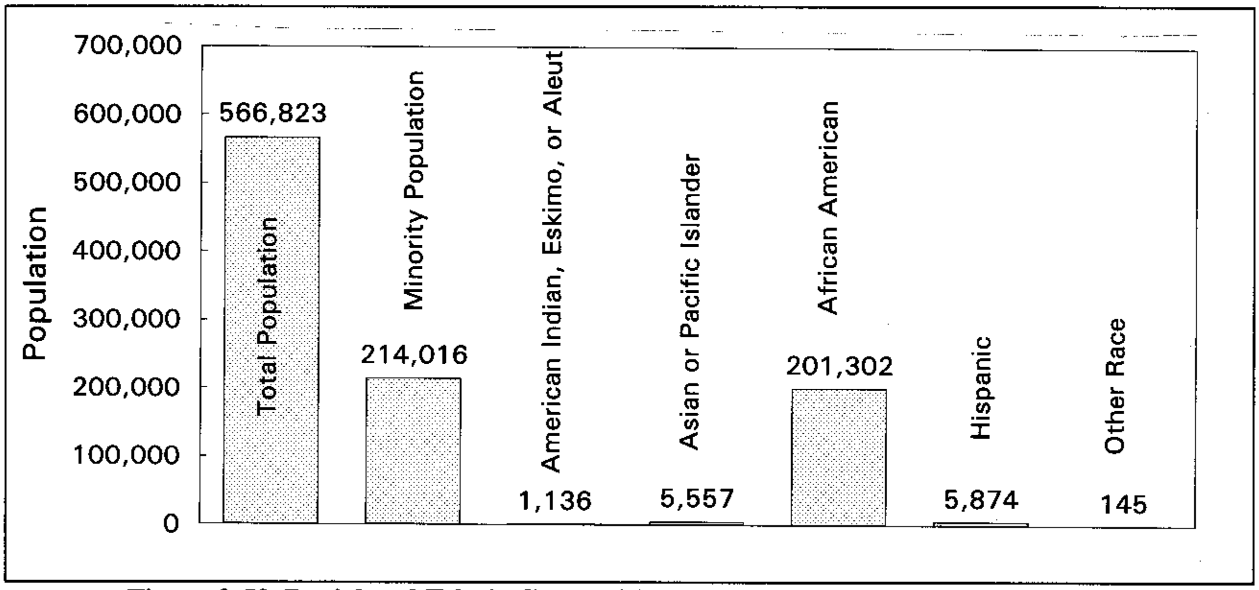


Figure 3-50 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Savannah River Site

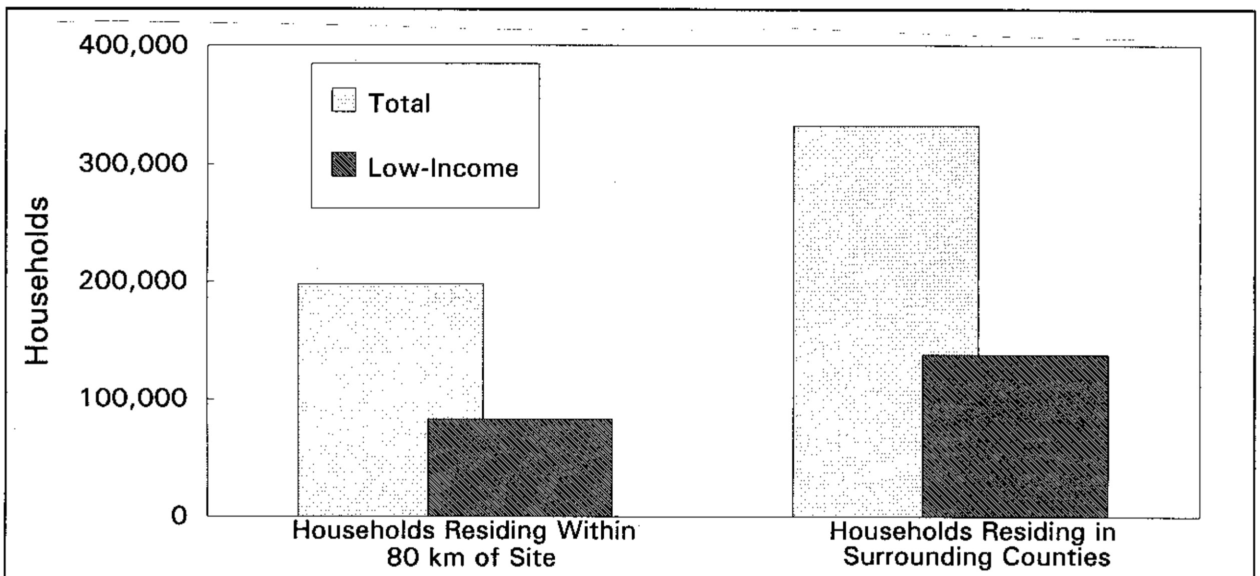


Figure 3-51 Low-Income Households Residing within 80 km (50 mi) of the Savannah River Site

3.3.1.10 Historical, Archaeological and Cultural Resources

By the end of Fiscal Year 1992, approximately 60 percent of the 800 km^2 (310 m^2) of the Savannah River Site had been examined, and 858 archaeological sites had been identified. Of these, 53 have been determined to be eligible for the National Register of Historic Places, and 650 have not been evaluated (DOE, 1995c).

Three Native American groups, the Yuchi Tribal Organization, the National Council of Muskogee Creek, and the Indian People's Muskogee Tribal Town Confederacy, have expressed concerns over sites and items of religious significance on the Savannah River Site.

Archaeologists have divided areas of the Savannah River Site into three sensitivity zones related to their potential for containing sites with multiple archaeological components or dense or diverse artifacts, and their potential for eligibility to the National Register of Historic Places (DOE, 1995c).

Zone One is the zone of highest archaeological site density, with a high probability of encountering large archaeological sites with dense and diverse artifacts, and high potential for nomination to the National Register of Historic Places. Zone Two covers areas of moderate archaeological site density that should contain sites of similar composition. Activities in this zone have a moderate probability of encountering archaeological sites, but a low probability of encountering large sites with more than three prehistoric components. All areas within the zone are conducive to site preservation. The zone has moderate potential for encountering sites that would be eligible for nomination to the National Register of Historic Places. Zone Three covers areas of low archaeological site density. Activities in this zone have a low probability of encountering archaeological sites and virtually no chance of encountering large sites with more than three prehistoric components. Therefore, potential for site preservation is low. Some exceptions to this definition have been discovered in Zone 3, so some sites in the zone could be considered eligible for nomination to the National Register of Historic Places.

3.3.2 Description of the Affected Environment at the Idaho National Engineering Laboratory

This section describes the potentially affected environment of the Idaho National Engineering Laboratory. The location of the site is shown in Figure 3-52.

3.3.2.1 Geology

The Idaho National Engineering Laboratory is located within a broad low-relief basin floored with basaltic lava flows and terrestrial sediments in the Eastern Snake River Plain physiographic province. The Snake River Plain extends in a broad arc from the Idaho-Oregon border in the west to the Yellowstone Plateau in the east, and contrasts sharply with the surrounding mountainous country of the Northern Basin and Range Province and the Idaho Batholith.

The Snake River Plain was formed in response to the movement of the North American Continent over a deep seated plume of hot mantle rocks. Movement of the continent and a northeast directed extension of the crust caused development of both the Eastern Snake River Plain and the northern Basin and Range Province over the past 17 million years. This movement has produced northwest trending normal faults in the Basin and Range Province and volcanic activity, including the formation of calderas, rhyolite domes, and volcanic rifts or vents.

Surface rocks on and near the Idaho National Engineering Laboratory are mostly an interlayered sequence of basalt flows and poorly consolidated sedimentary interbeds to a depth of 1 to 2 km (0.6 to 1.2 mi). Interbedded sediments are composed mostly of fine-grained silts and clays. A wide band of Quaternary

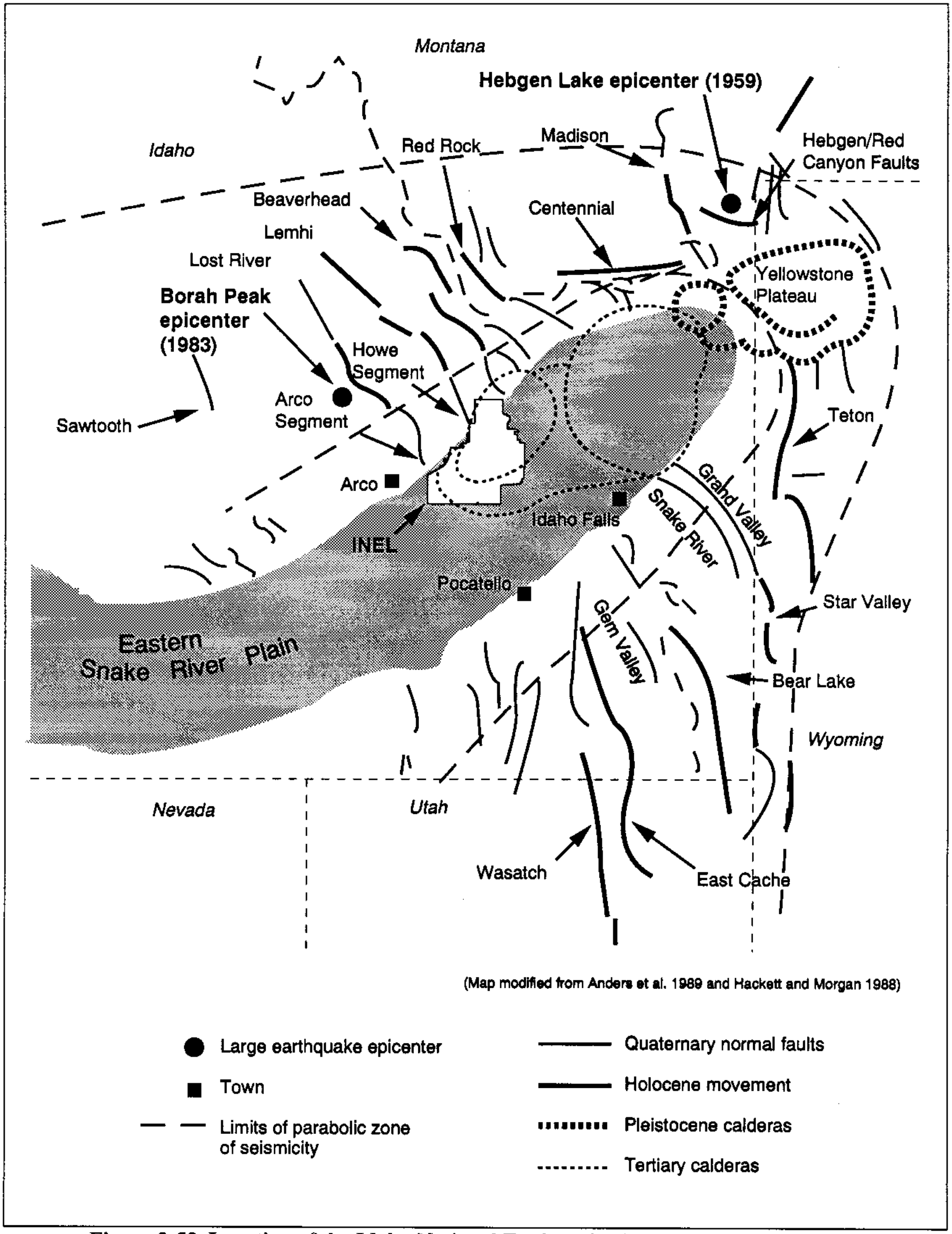


Figure 3-52 Location of the Idaho National Engineering Laboratory in Context of Regional Geologic Features

mainstream alluvium extends along the course of the Big Lost River (southwestern corner of the Idaho National Engineering Laboratory) to the Lost River Sinks area in the north-central portion of the Idaho National Engineering Laboratory. Lacustrine clay and sand deposits from the Ice Age Lake Terreton occur in the northern part of the site. Some of these beach sands were reworked by winds in late Pleistocene and Holocene times to form large dune fields in the northeastern part of the Idaho National Engineering Laboratory (Scott, 1982). Several Quaternary rhyolite domes occur along the Axial Volcanic Zone near the south and southeast borders of the Idaho National Engineering Laboratory. Paleozoic limestones, late-Tertiary rhyolitic volcanic rocks, and large alluvial fans occur in limited areas along the northwest margin of the site.

3.3.2.2 Seismology and Volcanology

The effusion of basaltic-lava flows from volcanic rift zones has been the predominant style of volcanism on the Eastern Snake River Plain, including the Idaho National Engineering Laboratory area, during the past 4 million years. Broad uplift of the ground surface, together with the opening of fissures and faults, accompany the ascending magma before eruptions. Vents for the basaltic volcanism are concentrated in northwest-trending volcanic rift zones and along the Axial Volcanic Zone (Kuntz, 1992).

Volcanic vents on the Eastern Snake River Plain are concentrated in several linear belts that trend northwest and northeast. These northwest-trending belts are associated with ground deformation features referred to as volcanic rift zones with fissures, faults, grabens, and monoclines. Volcanic vents are also concentrated in a northeast-trending zone along the axis of the Eastern Snake River Plain, called the Axial Volcanic Zone to distinguish it from volcanic rift zones (Hackett and Smith, 1992). Basaltic-lava eruptions appear to have been most frequent and most recent (approximately 5,200 years ago at Hell's Half Acre) along this zone, and at intersections of that zone with northwest-trending volcanic rift zones in the site area. Volcanic vents in this area are fed by northwest-trending dikes with seismicity associated with volcanism.

The Eastern Snake River Plain is surrounded by the seismically active Intermountain Seismic and Centennial Tectonic Belts (Smith and Arabasz, 1991). The Snake River Plain is devoid of earthquakes relative to the active areas surrounding it. The Eastern Snake River Plain has exhibited infrequently-occurring small magnitude (M 1.5) earthquakes.

3.3.2.3 Hydrology

3.3.2.3.1 Surface Water

Surface water at the Idaho National Engineering Laboratory accumulates into streams from local rainfall, snowmelt, and runoff originating in the mountain ranges located directly north and west of the Idaho National Engineering Laboratory. Surface water also includes water contained in man-made infiltration and evaporation ponds. Except for standing water in man-made ponds, there is little surface water at the Idaho National Engineering Laboratory. However, during wet years, when runoff from drainage basins or snowmelt is heavy, surface water bodies are formed.

The Idaho National Engineering Laboratory is located in Pioneer Basin, a closed drainage basin that includes three main surface water bodies: The Big Lost River, the Little Lost River, and Birch Creek. These sources of water drain mountain watersheds located north and west of the Idaho National Engineering Laboratory. However, most of the surface water flow is diverted for irrigation before it reaches site boundaries (Barraclough et al., 1981). This has resulted in little or no flow in these surface water bodies for several years within the boundaries of the Idaho National Engineering Laboratory (Pittman et al., 1988).

The Big Lost River is the major surface water body at the Idaho National Engineering Laboratory. The river flows between the Lost River Range and the Pioneer Mountains, draining approximately 3,755 km^2 (1,450 mi^2) of land before reaching the site. Approximately 48 km (30 mi) upstream of Arco, ID, Mackay Dam controls and regulates the flow of the river. During heavy runoff events, surface water from the Big Lost River is diverted to four spreading areas located south of the Idaho National Engineering Laboratory Diversion Dam. North of the diversion dam, the Big Lost River continues northward across the site to an area of natural infiltration basins (playas or sinks) near Test Area North. Flow within the Big Lost River channel may only reach a few kilometers southeast of Arco during drought years.

Birch Creek flows through an elongated, southeast-trending valley located between the Lemhi and Beaverhead Mountain Ranges, which cross the northwest corner of the Idaho National Engineering Laboratory to Playa 4 near Test Area North. Birch Creek drains an area of approximately 1,943 km^2 (750 mi^2). In the summer, upstream of the Idaho National Engineering Laboratory, surface water from Birch Creek is diverted for irrigation and to produce hydropower. In the winter, water flows in a man-made channel constructed 6.4 km (4 mi) north of Test Area North, where it infiltrates into channel gravel.

The Little Lost River drains the slopes of the Lemhi and Lost River Ranges, an area of approximately 1,826 km^2 (705 mi^2). Surface water from the Little Lost River has not reached the Idaho National Engineering Laboratory in recent times; during high streamflow years, however, water will reach the site and infiltrate into the subsurface (EG&G Idaho Inc., 1984).

Surface water generated from local precipitation will flow into topographic depressions (lower elevations than the surrounding terrain). Surface water within the depressions either evaporates or infiltrates into the ground. Localized flooding can occur at the Idaho National Engineering Laboratory when the ground is frozen and melting snow is combined with heavy spring rains. Other facility areas have been threatened by flooding because of ice jams in the Big Lost River and its diversion channel (McKinney, 1985).

Intermittent surface flow and the Idaho National Engineering Laboratory Diversion Dam have effectively prevented the Big Lost River from flooding onto the site. Flood plains in existence before the diversion dam and channels were built are not active. Based on historical data from past storm events, if heavy runoff occurs and surface water flow from the Big Lost River is diverted to the four spreading areas, water will not overflow the banks of the existing Big Lost River channel onsite during 100- and 500-yr floods (floods that occur on an average of every 100 or 500 years).

Onsite flooding from the Big Lost River may occur if high water in the MacKay Dam or the Big Lost River is coupled with a dam failure. Assuming the Mackay Dam fails, significantly higher flows and greater flooding would be associated with the probable maximum flood than either the 100- or 500-year floods (Figure 3-53).

Surface Water Quality: Water quality in the Big Lost River, Little Lost River, and Birch Creek is similar, and has not varied significantly over the period of record. The chemical composition of these water bodies is determined primarily by the mineral composition of the rocks in surrounding mountain ranges northwest of the site, and by the chemical composition of irrigation water in contact with the surface water

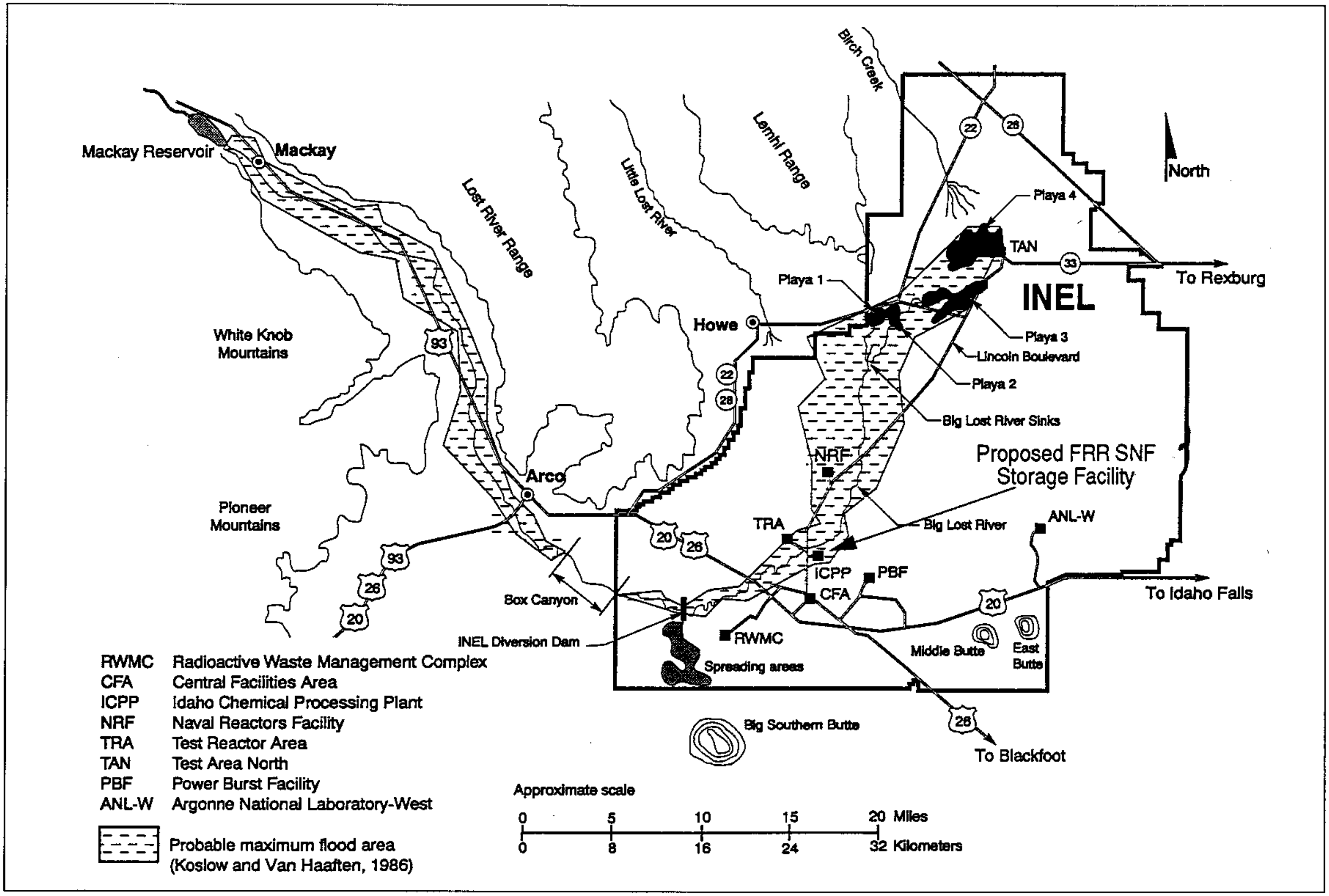


Figure 3-53 Selected Facilities and Predicted Inundation Map for Probable Maximum Flood-Induced Overtopping Failure of Mackay Dam at the Idaho National Engineering Laboratory

(Robertson et al., 1974; Bennett, 1990). Chemical and physical parameters measured in the Big Lost River, Little Lost River, and Birch Creek do not exceed drinking water quality standards for the parameter analyzed (DOE, 1995c).

The Idaho National Engineering Laboratory activities do not directly affect the quality of surface water outside the site, because discharges from the Idaho National Engineering Laboratory facilities are made to man-made seepage and evaporation basins or stormwater injection wells. Surface water is not withdrawn by humans for use at the Idaho National Engineering Laboratory, and discharge of effluents to natural surface waters does not occur. In addition, surface water does not flow directly offsite (Hoff et al., 1990). However, water from the Big Lost River, as well as from seepage of evaporation basins and stormwater injection wells, does infiltrate into the Snake River Plain Aquifer, and will eventually reach the Snake River (Robertson et al., 1974; Wood and Law, 1988; Bennett, 1990).

3.3.2.3.2 Groundwater

Groundwater at the site occurs in the Snake River Plain Aquifer and the vadose zone. The Idaho National Engineering Laboratory overlies the Snake River Plain Aquifer, which covers an area of approximately 24,900 km^2 (9,614 mi^2). Groundwater in the aquifer generally flows south and southwestward across the Snake River Plain, discharging into the Snake River at the Thousand Springs Area near Twin Falls, ID.

The Snake River Plain Aquifer is recharged by seepage of irrigation water, stream channel and canal leakage, tributary drainage basin underflow, and direct infiltration from precipitation (Garabedian, 1989). The drainage basin recharging the aquifer covers an area of approximately 90,643 km^2 (35,000 mi^2). Most recharge occurs in surface water-irrigated areas and along the northeastern margins of the plain. A majority of the groundwater discharged from the aquifer is through springs that flow into the Snake River, and through pumping for irrigation purposes. Major springs and seepages that flow from the aquifer are located near the American Falls Reservoir (southwest of Pocatello) and the Thousand Springs area between Milner Dam and King Hill (near Twin Falls, ID).

Water storage in the Snake River Plain Aquifer is estimated at 2.5 x 10^{+12}m^3 (8.8 x 10^{+13}ft^3) (Robertson et al., 1974). Irrigation wells yield as much as 26.5 m^3 per min (7,000 gal per min) of water (Garabedian, 1989). Groundwater discharges primarily from the aquifer through springs that flow into the Snake River and from pumping for irrigation purposes. Major springs and seepages that flow from the aquifer are localized near the American Falls Reservoir (southwest of Pocatello) and the Thousand Springs area between Milner Dam and King Hill.

The Idaho National Engineering Laboratory covers 2,305 km^2 (890 mi^2) of the north-central portion of the Snake River Plain Aquifer. Most of the aquifer is composed of a thick sequence of relatively thin basaltic lava flows with sedimentary interbeds extending to depths greater than 1,067 m (3,500 ft) below the land surface (Irving, 1993). The basalt flows are interbedded with sedimentary layers formed during periods between volcanic eruptions when the basalt was exposed and sediments collected on the land surface. A majority of the groundwater moves horizontally through fractured, basaltic interflow zones (broken and rubble zones) that occur at various depths. Water also moves vertically along joints and the interfingering edges of interflow zones (Garabedian, 1986). Sedimentary interbeds restrict the vertical movement of groundwater.

Depths to the water table from the land surface at the Idaho National Engineering Laboratory range from approximately 61 m (200 ft) in the north, to more than 274 m (900 ft) in the south (Pittman et al., 1988). The upper surface of the aquifer is unconfined over most of its extent. However, the aquifer generally behaves as if it were partially confined, because water moves through dense basalt with interbedded sediments and water-bearing basaltic fracture zones at different rates. The base of the aquifer coincides with the tip of a thick, widespread sequence of clay, silt, sand, and basalt that occurs at depths ranging from 244 to 457 m (800 to 1,500 ft) below the land surface. The thickness of the aquifer is primarily controlled by the geologic setting, and therefore varies across the Idaho National Engineering Laboratory (Anderson, 1991).

The rate water moves through the aquifer depends on the gradient (change in elevation with distance) of the water table, the porosity of the soil and bedrock (void spaces in aquifer materials), and the hydraulic conductivity of the soil and bedrock (capacity of a porous media to transport water). Across the Idaho National Engineering Laboratory, the horizontal gradient of the water table ranges from 0.19 to 2.9 m/km (1 to 15 ft/mi) (Ackerman, 1991). Vertical hydraulic gradients (change in elevation, pressure, and velocity with distance in a given direction) are usually less than 0.01 m/m (0.01 ft/ft) in the first 61 m (200 ft) below the land surface, and less than 0.02 m/m (0.02 ft/ft) in the first 168 m (550 ft) of the saturated thickness. Groundwater flows horizontally at velocities ranging from 1.5 to 7.6 m/day (5 to 25 ft/day). However, most of the water flows from 1.5 to 3 m/day (5 to 10 ft/day) (Robertson et al., 1974).

Transmissivity values vary widely across the Idaho National Engineering Laboratory. Data from 183 single-well tests at 94 wells provide estimates of transmissivity ranging from 0.1 to 70,604 m^2 per day (1.1 to 760,000 ft^2 per day) (Ackerman, 1991). The lowest transmissivity rates are generally at the northern end of the Idaho National Engineering Laboratory, and the highest rates are near the Test Reactor Area in the south-central part of the Idaho National Engineering Laboratory. Aquifer storativity is estimated using results from multiple-well aquifer tests. Values of storativity (or storage coefficients) range from 0.01 to 0.06, indicating generally unconfined aquifer conditions at the site. The aquifer behaves as an unconfined system, which increases well yields. However, clay layers and dense, unfractured basalts in the aquifer are locally confining.

Since aquifer porosity and hydraulic conductivity decrease with depth, most of the water in the aquifer moves through the upper 61 to 152 m (200 to 500 ft) of the Quaternary basalts. Estimated flow rates within the aquifer range from 1.5 to 6.1 m (5 to 20 ft) per day. Recharge to the aquifer near the Idaho National Engineering Laboratory originates from precipitation in the mountains to the north and west. Lesser amounts of recharge occur from local precipitation and snowmelt (Barraclough et al., 1981).

The vadose zone at the Idaho National Engineering Laboratory extends from the land surface down to the water table, with a thickness ranging from 61 m (200 ft) in the north, to greater than 274 m (900 ft) in the south. The vadose zone consists of surface sediments and relatively thin, basaltic lava flows with occasional interbedded sediments. The surface sediments are composed of clay, silt, sand, and gravel. Thick surficial deposits of (primarily) clay and silt are found in the northern portion of the Idaho National Engineering Laboratory, and thin deposits are found southward, where basalt is exposed at the surface.

Perched water at the site generally occurs under presence of disposal ponds or other surface water features. Perched water bodies have been detected at the Idaho Chemical Processing Plant, Test Reactor Area, Test Area North, and Radioactive Waste Management Complex facility areas.

Groundwater Quality: Previous waste discharges to unlined ponds and deep wells have introduced radionuclides, nonradioactive metals, inorganic salts, and organic compounds to the subsurface (DOE, 1995c).

Radionuclide concentrations in the Snake River Plain Aquifer beneath the Idaho National Engineering Laboratory have decreased since the mid-1980's because of changes in disposal practices, radioactive decay, adhesion of radionuclides to rocks and minerals, and dilution by natural surface water and groundwater entering the aquifer (Pittman et al., 1988; Orr and Cecil, 1991). Radionuclides released and observed in the soil and groundwater include tritium, strontium-90, iodine-129, cobalt-60, cesium-137, plutonium-238, plutonium-239/240, and americium-241 (Golder Associates, 1993).

The Idaho National Engineering Laboratory has released sodium, chromium, lead, and mercury on the site and into the subsurface through unlined ponds and deep wells. Sodium is the greatest quantity of material released from the waste treatment processes. However, sodium is not toxic. In 1988, chromium concentrations exceeding the maximum allowable contaminant level were measured near the Test Reactor Area. Lead and mercury have occurred at concentrations below the maximum contaminant level near the Idaho Chemical Processing Plant (Orr and Cecil, 1991).

Concentrations of volatile organic compounds have been detected in the aquifer beneath the Idaho National Engineering Laboratory. Concentrations of the following compounds exceeding the maximum contaminant levels have occurred in and near the Test Area North disposal well: Carbon tetrachloride; chloroform; 1,2-cis-dichloroethylene; 1,1-dichloroethylene; 1,2-trans-dichloroethylene; trichloroethylene; tetrachloroethylene; and vinyl chloride (Leenheer and Bagby, 1982; Mann and Knobel, 1987; Liszewski and Mann, 1993).

Groundwater uses on the Snake River Plain include irrigation, food processing and agriculture, and domestic, rural, public, and livestock supply. Water use for the upper Snake River drainage basin and Snake River Plain Aquifer was 16.4 x 10^{+10} cubic m (4.3 x 10^{+12} gal) per year during 1985, which was more than 50 percent of the water used in Idaho, and approximately 7 percent of agricultural withdrawals in the nation. Site activities withdraw water at an average rate of 7,400,000 m^3 (1.9 x 10^{+9} gal) per year. This rate is equal to approximately 0.4 percent of the water consumed in the Eastern Snake River Plain Aquifer (DOE, 1995c).

Since groundwater supplies 100 percent of the drinking water consumed within the Eastern Snake River Plain (Gaia Northwest, 1988), and an alternative drinking water source or combination of sources is not available, the U.S. Environmental Protection Agency designated the Snake River Plain Aquifer a sole-source aquifer in 1991 (EPA, 1991b).

The Idaho Department of Water Resources manages groundwater resources to meet the State's water needs. Idaho operates under the system of appropriation rights — the senior appropriation has priority in times of shortage, or "first in time, first in right."

DOE holds a Federal Reserved Water Right for the Idaho National Engineering Laboratory, which permits a water pumping capacity of 2.3 m^3 per sec (80 ft^3 per sec), and a maximum water consumption of 43 million m^3 per year (11.4 billion gal per year) for drinking, process water, and noncontact cooling. Because it is a Federal Reserved Water Right, the Idaho National Engineering Laboratory's priority on water rights dates back to the establishment of the Idaho National Engineering Laboratory.

3.3.2.4 Meteorology

The Eastern Snake River Plain climate exhibits low relative humidity, wide daily temperature swings, and large variations in annual precipitation. Several topographic characteristics of the area (including altitude, latitude, and intermountain setting) influence the climate of the site area.

Wind: The terrain impacts the regional wind flow. The Idaho National Engineering Laboratory is in the belt of prevailing westerlies. Nighttime winds are common at the Idaho National Engineering Laboratory on clear or partly cloudy nights. These winds blow from the northeast and are formed by the rapid rotational cooling of the near-ground air layer on mountain slopes. Wind directions with the highest frequency of occurrence as measured onsite, are from south to west-southwest and from northwest to northeast (Clawson et al., 1989).

The highest hourly average near-ground (6.1 m, 20 ft level) windspeed measured onsite is 22.8 m per sec (51 mph) from the west-southwest. The maximum instantaneous gust at this level was 34.9 m per sec (78 mph). Some of the highest windspeeds occur in the spring, which coincides with strong prevailing westerlies at higher altitudes (Clawson et al., 1989). Apart from thunderstorms, severe weather is uncommon. Visibility in the region is good due to the low moisture content of the air and minimal sources of visibility-reducing pollutants. The background visibility is estimated at 60 km (37.5 mi).

Temperature and Humidity: Average seasonal temperatures measured onsite range from -7.3°C (18.8°F) in winter to 18.2°C (64.8°F) in summer, with an annual average temperature of about 5.6°C (42°F). Temperature extremes measured at the Idaho National Engineering Laboratory range from a summertime maximum of 39.4°C (103°F), to a wintertime minimum of -45°C (-49°F). Temperature differences in excess of 13.9°C (25°F) have been observed onsite (Clawson et al., 1989).

The annual average relative humidity at the Idaho National Engineering Laboratory is 50 percent, with monthly average maximum values ranging from 59 percent in July to 89 percent in February and December, and with monthly average minimum values ranging from 16 percent in June and July to 47 percent in January (Clawson et al., 1989).

Precipitation: Annual precipitation is light, averaging 22.12 cm (8.71 in). The monthly average precipitation peaks in May and June. For the rest of the year, precipitation is uniformly distributed, averaging 1.27 to 1.78 cm (0.5 to 0.7 in) monthly. The highest monthly average snowfall of 16.26 cm (6.4 in) occurs in December, with similar amounts during January. The average annual snowfall is 70.1 cm (27.6 in). Maximum annual snowfall is 151.6 cm (59.7 in), and minimum annual snowfall is 17.3 cm (6.8 in) (Clawson et al., 1989).

Atmospheric Dispersion: Vertical diffusion of pollutants may be restricted or enhanced by the vertical temperature gradient of the atmosphere (that is, lapse or inversion conditions). These inversions are often ground-based, meaning that the temperature increases with height from the ground (Clawson et al., 1989).

Air Quality: The population of the Eastern Snake River Plain is exposed to environmental radiation from both natural and man-made sources.

Background radiation includes sources such as cosmic rays, radioactivity naturally present in soil, rocks, and the human body, and airborne radionuclides of natural origin (such as radon). Radioactivity still remaining in the environment as a result of atmospheric testing of nuclear weapons also contributes to the background radiation level, although in very small amounts.

The Programmatic SNF&INEL Final EIS presents a summary of the estimated natural background dose by exposure source for residents of the Eastern Snake River Plain (DOE, 1995c). The cumulative annual dose, 351 mrem, is caused largely by the inhalation of radioactive particles formed by the decay of naturally occurring radon.

Sources of radiological emissions at the Idaho National Engineering Laboratory result from operation of research and training reactors, spent nuclear fuel testing and processing, irradiated material and fuel examination, nuclear waste treatment and storage, and depleted uranium armor production. These operations can result in the release of radioactivity to air, either directly (e.g., through stacks or vents) or indirectly (e.g., by re-suspension of radioactivity on contaminated grounds). Concentrations of radionuclides in direct releases are monitored or estimated based on knowledge of the materials used and activities performed. Indirect releases are estimated using engineering calculations that relate surface contamination levels to expected airborne concentrations.

Table 4.7-2 of Appendix B, Volume 1 of the Programmatic SNF&INEL Final EIS provides a summary of the principal types of airborne radioactivity emitted from the Idaho National Engineering Laboratory operations during 1991 (DOE, 1995c). The information summarized provides a perspective on the general nature and magnitude of airborne radiological emissions from current the Idaho National Engineering Laboratory operations. These emissions include the noble gases (argon, krypton, and xenon) and iodine; particulate fission products such as rubidium, strontium, and cesium; radionuclides formed by neutron activation such as tritium (hydrogen-3), carbon-14, and cobalt-60; and very small quantities of heavy elements such as uranium, thorium, plutonium, and their decay products. The emissions listed are considered representative of a maximum baseline year. The radionuclide with the highest emission rate is the noble gas, krypton-85. Most of the krypton-85 emissions result from the chemical reprocessing of spent nuclear fuel at the Idaho Chemical Processing Plant.

3.3.2.5 Ecology

The Idaho National Engineering Laboratory vegetation consists primarily of shrub-steppe vegetation, and is a small fraction of the 45 million ha (112.5 million acres) of this vegetation type found in the Intermountain West. Vegetation communities range from shadscale-steppe vegetation at lower altitudes, through sagebrush and grass-dominated communities, to juniper woodlands along the foothills of the nearby mountains and buttes. Big sagebrush and rabbitbrush are the most common and noticeable shrub species on the Idaho National Engineering Laboratory. Other common shrubs include winterfat, perennial broomweed, black sage, and saltbrush species. Grass species common to the Idaho National Engineering Laboratory include Indian ricegrass, Great Basin wildrye, needle-and-threadgrass, wheatgrasses, bottlebrush squirreltail, and cheatgrass. Common flora species include mustards, summer cypress, and Russian thistle. Fifteen vegetation classes have been identified at the Idaho National Engineering Laboratory. The classes can be grouped into six major types: juniper woodland, native grassland, shrub-steppe, lava, modified, and wetland vegetation types.

The Idaho National Engineering Laboratory supports animal communities typical of Great Basin high desert vegetation and habitats. About 270 vertebrate species have been observed, including 46 mammal, 204 bird, 10 reptile, 2 amphibian, and 9 fish species (Arthur et al., 1984; DOE, 1995c). Thirty-seven mammal species or their signs have been observed at the site. Included are 14 rodents, 4 lagomorphs (rabbits and hares), 6 bats, 6 carnivores, 4 ungulates (hoofed animals), and 1 shrew. A total of 185 bird species were recorded on the site (DOE, 1995c). Other species may be present on the Idaho National Engineering Laboratory since more than 216 bird species have been reported in southeastern Idaho in similar habitats. Of these species, 32 are game birds, 26 are waterfowl, 82 are passerines, and 22 are raptors (for example, hawks, eagles, and owls). Waterfowl use man-made ponds and lagoons at the Idaho National Engineering Laboratory, and the Idaho National Engineering Laboratory is a nesting and wintering area for raptors. Sage grouse and their habitat are found throughout the site. Ten reptile and one amphibian species have been observed on the Idaho National Engineering Laboratory, but only five are common or abundant (DOE, 1995c). These species include western rattlesnake, short-horned lizard, sagebrush lizard, Great Basin spadefoot toad, and western garter snake.

Two Federal endangered and six Federal Category 2 Candidate animal species were identified by the U.S. Fish and Wildlife Service as potentially existing on the Idaho National Engineering Laboratory (DOE, 1995c). Seven additional animal species listed by the State as species-of-special-concern were identified. No Federal-or State-listed plant species were identified as potentially existing on the Idaho National

Engineering Laboratory. The bald eagle and peregrine falcon are Federally listed endangered species. The bald eagle is a winter resident and is locally common in the far north end of the Idaho National Engineering Laboratory, and on the western edge of the site near Howe. Peregrine falcons have been rarely observed in the winter. Neither species is known to nest on the site, and neither species is commonly observed near facilities (Reynolds, 1993).

The Candidate Category 2 species identified by the U.S. Fish and Wildlife Service as potentially existing at the Idaho National Engineering Laboratory include four birds and two mammals. Both the long-billed curlew and the white-faced ibis are uncommon migrants at the Idaho National Engineering Laboratory, associated with aquatic and riparian habitats. Swainson's hawk and ferruginous hawk nest on and migrate through the site. These species are found throughout the site, but are observed more frequently in the juniper woodlands and riparian areas where they nest. The loggerhead shrike, which uses shrub-steppe vegetation to nest, is also found on the Idaho National Engineering Laboratory. Breeding and hibernation caves for Townsend's big-eared bat have been observed on the Idaho National Engineering Laboratory. About six caves are used, the nearest being in excess of 7 km (3 mi) from the nearest facility. The pygmy rabbit is common on the Idaho National Engineering Laboratory (DOE, 1995c). The habitat for this species is grass and shrub communities found throughout much of the Idaho National Engineering Laboratory.

State species-of-special-concern include three aquatic, one raptor, and three bat species. The aquatic bird species are rare migrants (DOE, 1995c). The gray falcon has only been observed a few times. Of the bats, only the California myotis has been observed, and it is rare. Ten plant species identified by other Federal agencies (U.S. Bureau of Land Management and U.S. Forest Service) and the Idaho Native Plant Society as sensitive, rare, or unique are known to occur on the Idaho National Engineering Laboratory (Chowlewa and Henderson, 1984). These species are not, however, protected under State or Federal laws. Most of these species are not found near any the Idaho National Engineering Laboratory facilities, and are uncommon on the Idaho National Engineering Laboratory because they require unique microhabitat conditions.

Potential wetlands are found throughout the Idaho National Engineering Laboratory and cover about 0.25 percent (3,322 ha or 8,206 acres) of the site. The wetlands are associated primarily with drainages, although some wetlands are associated with natural playas, man-made ponds, and drainage ditches. More than 70 percent of the potential wetlands are identified as palustrine (marsh lands). These wetlands are found mainly near the Big Lost River and its spreading areas and playas, the Birch Creek Playa, and an area to the north and in the general vicinity of Argonne National Laboratory-West. Potential lacustrine wetlands are found in the central and south central portions of the site. Potential riverine wetlands were mapped by the U.S. Fish and Wildlife Service near the Big Lost River, Birch Creek, and Big Lost River Playas. Man-made wetlands include industrial waste and sewage treatment ponds, borrow pits, and gravel pits associated with site facilities. Limited riparian communities with large trees are found along the drainages of the Big Lost River, and very limited riparian habitat is located on Birch Creek.

3.3.2.6 Land Use

The Idaho National Engineering Laboratory encompasses 231,049 ha (570,934 acres). Only about two percent of the land (4,614 ha or 11,400 acres) is used for facilities and operations supporting energy research and development and waste management. In addition to industrial and support land uses associated with each of the facility areas at the Idaho National Engineering Laboratory, other land uses exist within the Idaho National Engineering Laboratory site boundaries.

Approximately five percent of the total the Idaho National Engineering Laboratory site area, or 13,349 ha (33,000 acres), is devoted to public road, utility, and railway rights-of-way crossing the site (DOE, 1995c). Rights-of-way at the Idaho National Engineering Laboratory are granted and administered by the U.S. Department of the Interior and the Bureau of Land Management.

More than half of the Idaho National Engineering Laboratory is used for grazing. The actual amount of the Idaho National Engineering Laboratory land used for grazing varies from year to year, but is usually between 121,410 and 141,645 ha (300,000 and 350,000 acres). In 1992 for example, 121,853 ha (301,094 acres) were grazed (DOE, 1995c). Grazing is not allowed within 3.2 km (2 mi) of any nuclear facility, and dairy cattle are not permitted on the site to avoid the possibility of milk contamination by long-lived radionuclides. The U.S. Sheep Experiment Station, located approximately 42.6 km (26.5 mi) northeast of the site, uses a 364 ha (900 acre) portion of the Idaho National Engineering Laboratory for a winter feed lot for approximately 5,000 sheep.

The Idaho National Engineering Laboratory also supports periodic uses associated with onsite resources. Two sites within the Idaho National Engineering Laboratory site boundary are listed in the National Register of Historic Places: Experimental Breeder Reactor-I, designated in 1966, and Goodale's Cutoff, a portion of the Oregon Trail that crosses the southwest corner of the Idaho National Engineering Laboratory, designated in 1974. In addition to public tours, the Idaho National Engineering Laboratory occasionally supports controlled hunting within the site boundaries. Each year the Idaho Department of Fish and Game and DOE jointly determine whether or not to allow controlled hunts of wild game populations living on the Idaho National Engineering Laboratory property (DOE, 1995c).

Several uses associated with the National Environmental Research Park designation occur at the Idaho National Engineering Laboratory. The Idaho National Engineering Laboratory's cool desert ecosystem provides a controlled outdoor laboratory where scientists can study changes to the natural environment caused by human activities.

3.3.2.7 Noise

Sources of man-made noise at the Idaho National Engineering Laboratory include noise from operation and construction activities; bus, car, and truck traffic; aircraft; security force training exercises; and the Idaho National Engineering Laboratory railroad. Previous studies have assessed noise impacts of existing the Idaho National Engineering Laboratory operational activities (Abbott et al., 1990). These studies concluded that because of the remote location of the Idaho National Engineering Laboratory, there are no known noise conditions associated with existing onsite operations that adversely affect individuals at offsite locations. Studies of the noise effects on wildlife at the Kennedy Space Center show that even very high noise levels have little significance on wildlife productivity (Leonard, 1993).

3.3.2.8 Transportation

Roads are the primary access to and from the site, and workers are transported primarily by bus and private vehicles. Commercial shipments are transported by road and air, and some bulk materials are transported by rail. Waste is transported by road and rail. The existing regional highway system and site roadways are shown in Figure 3-54. To maintain a supply of goods and services and to transport workers to the Idaho National Engineering Laboratory, an onsite road system of approximately 145 km (90 mi) of paved surface has been developed. About 29 km (18 mi) of this network are considered service roads and are closed to the public. In addition to the site facilities, DOE owns or leases office and technical buildings throughout Idaho Falls for approximately 4,000 DOE and DOE contractor personnel to administer and support work at the Idaho National Engineering Laboratory. DOE shuttle vans provide

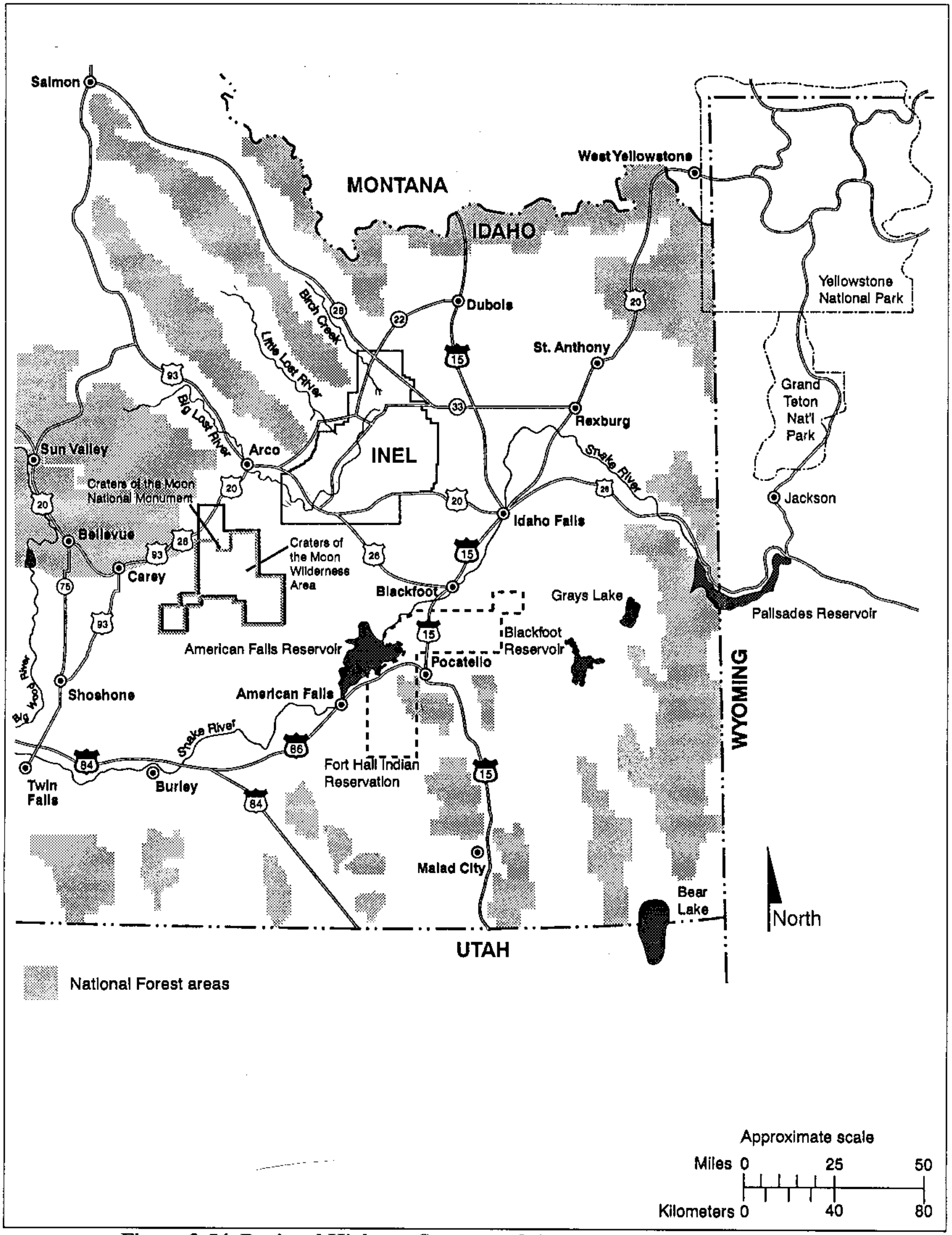


Figure 3-54 Regional Highway System and the Idaho National Engineering Laboratory Site Roadways

hourly transport between in-town facilities. Four major modes of transit use the regional highways, community streets, and site roads to transport people and commodities: DOE buses and shuttle vans, DOE motor pool vehicles, commercial trucks, and private automobiles.

Idaho Falls receives railroad freight service from Butte, Montana to the north, and from Pocatello, and Salt Lake City, Utah to the south via Union Pacific. The Union Pacific Railroad's Mackay Branch, which crosses the southern portion of the Idaho National Engineering Laboratory, provides rail service to the Idaho National Engineering Laboratory. This branch connects with a DOE-owned spur line at Scoville Siding, and then links with developed areas within the Idaho National Engineering Laboratory. Rail shipments to and from the Idaho National Engineering Laboratory are usually limited to bulk commodities, spent nuclear fuel, and radioactive materials.

One major airline carrier, Delta Airlines, provides Idaho Falls with jet aircraft passenger and cargo service. Two other air carriers, Horizon and Skywest, provide commuter service. Daily service includes flights to and from Boise and Salt Lake City. Landings in Idaho Falls for 1991 and 1992 totalled 5,367 and 5,578, respectively. The combined number of passengers leaving and arriving at Idaho Falls and Pocatello for 1991 and 1992 was 282,185 and 285,047, respectively. Non-DOE air traffic over the Idaho National Engineering Laboratory is restricted, and non-DOE aircraft are not permitted to use the site.

From 1987 through 1992, the average motor vehicle accident rate was 0.94 accident per million km (1.5 accidents per million mi) for Idaho National Engineering Laboratory vehicles, which compares with an accident rate of 1.5 accidents per million km (2.4 accidents per million mi) for all DOE complex vehicles and 8 accidents per million km (12.8 accidents per million mi) nationwide for all motor vehicles. There are no recorded rail or air accidents associated with the Idaho National Engineering Laboratory and, to date, no fatal air traffic accidents have involved flights through either the Idaho Falls or Pocatello airports (DOE, 1995c). Spent nuclear fuel and radioactive, hazardous, industrial, municipal, and recyclable wastes are transported at the Idaho National Engineering Laboratory.

3.3.2.9 Socioeconomics

In general, population growth in the region has mirrored population growth in Idaho as a whole over the past 30 years (Figure 3-55). Although growth was not evenly distributed among the counties, total regional population increased by 47 percent between the years 1960 to 1990, which is comparable to a Statewide growth rate of 51 percent. During this period, Madison County experienced the most rapid growth in the region, its population increasing by more than 150 percent. By contrast, the two smallest counties in the region, Butte and Clark, actually lost population between the years 1960 and 1990. The largest increases in the population of Idaho Falls occurred during the 1950's, which can be partially explained by the formation and the development of the National Reactor Testing Station, which evolved into the Idaho National Engineering Laboratory. By contrast, Pocatello grew primarily in the 1940's and the 1960's. Idaho Falls has continued to grow, increasing in population by 11 percent from the years 1980 to 1990. During the same period, however, Pocatello's population declined by 0.7 percent.

Historically, the economy of the seven-county region relied predominantly on natural resource use and extraction. To this day, farming, ranching, and mining remain important components of the regional economy. Almost all manufacturing in the region is a form of food processing, and mining or mineral processing. Idaho Falls is a regional retail and service center for southeastern Idaho. Similarly, Pocatello has evolved into an important processing and distribution center for the surrounding agricultural areas, and is a regional center for higher education. Retail trade and educational services are the two largest regional employment categories, providing 17.6 and 11.4 percent of the total employment in the region. Tourism is also considered to be an important component of the regional economy. The economy of the seven-county

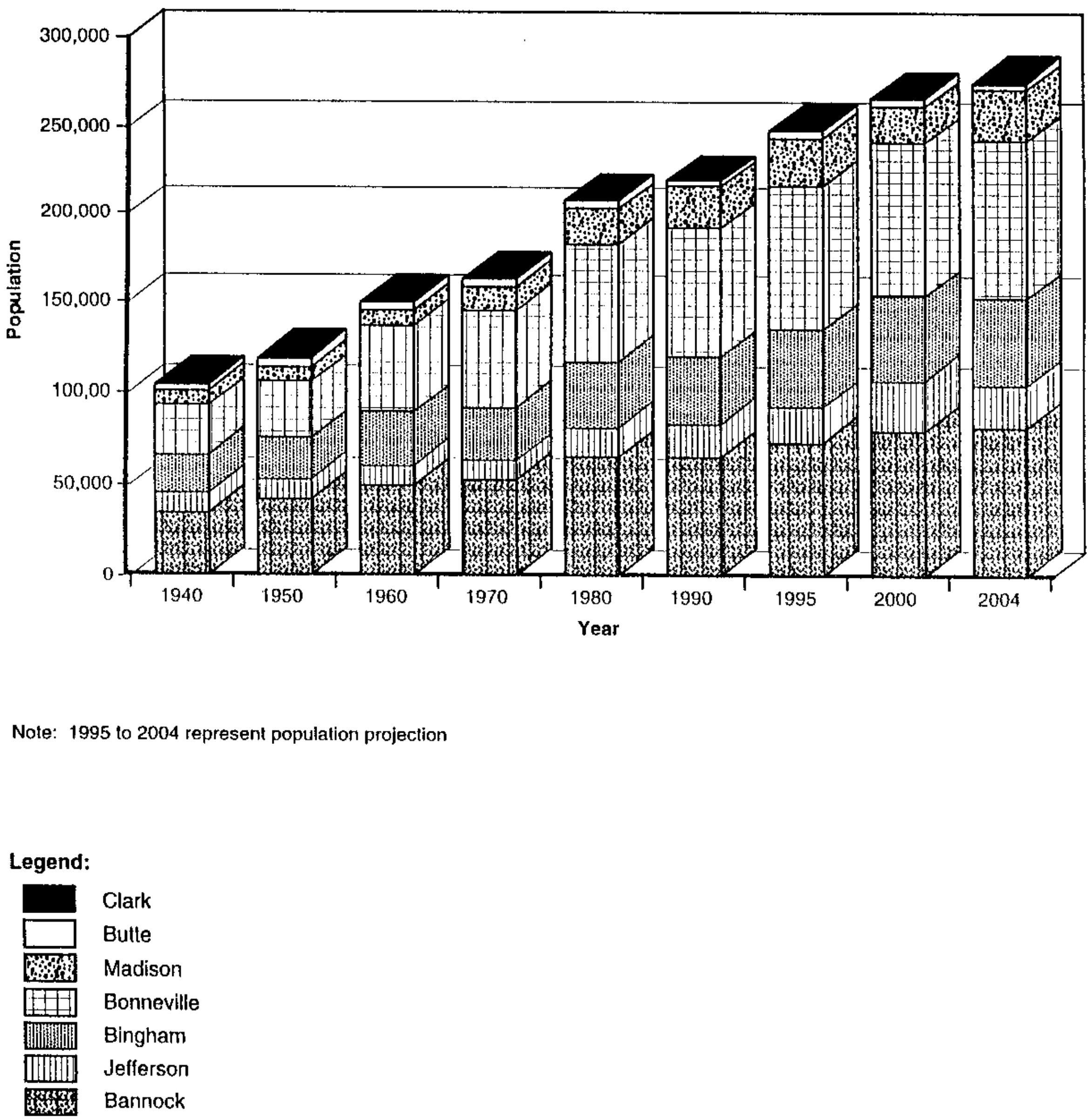


Figure 3-55 Actual and Projected Total Population for the Seven-County Region Surrounding the Idaho National Engineering Laboratory for the Years 1940 through 2004

region, however, currently revolves around agriculture and the Idaho National Engineering Laboratory (the single largest employer in the seven-county region). As of January 1991, DOE and its contractors employed more than 12,425 persons, with an estimated annual payroll of $440 million. Annual funding for the Idaho National Engineering Laboratory in 1991 was more than $1.1 billion.

As of January 1992, 12,803 contractor and Government personnel were employed at the Idaho National Engineering Laboratory, representing about 12 percent of the total available jobs in the seven-county region. Total employment at the Idaho National Engineering Laboratory has increased by more than 23 percent in the past 4 years. The Idaho National Engineering Laboratory has a large influence on both the regional economy and the economy of Idaho. During Fiscal Year 1992, total expenditures at the Idaho National Engineering Laboratory directly and indirectly supported 36,395 jobs, and generated an estimated $945 million in total earnings within the seven-county region.

Bonneville and Bannock Counties are the focal points of the housing market. These two counties provide 67 percent of the total housing in the region. A shortage of single-family housing presently exists near Idaho Falls because construction has not kept up with increasing demand. Total population living within the 80 km (50 mi) radius around the site is about 176,311. The ethnic composition of this population is presented in Figure 3-56. The number of low-income households within the same area of influence is presented in Figure 3-57. Approximately 40 percent of these households are low income families.

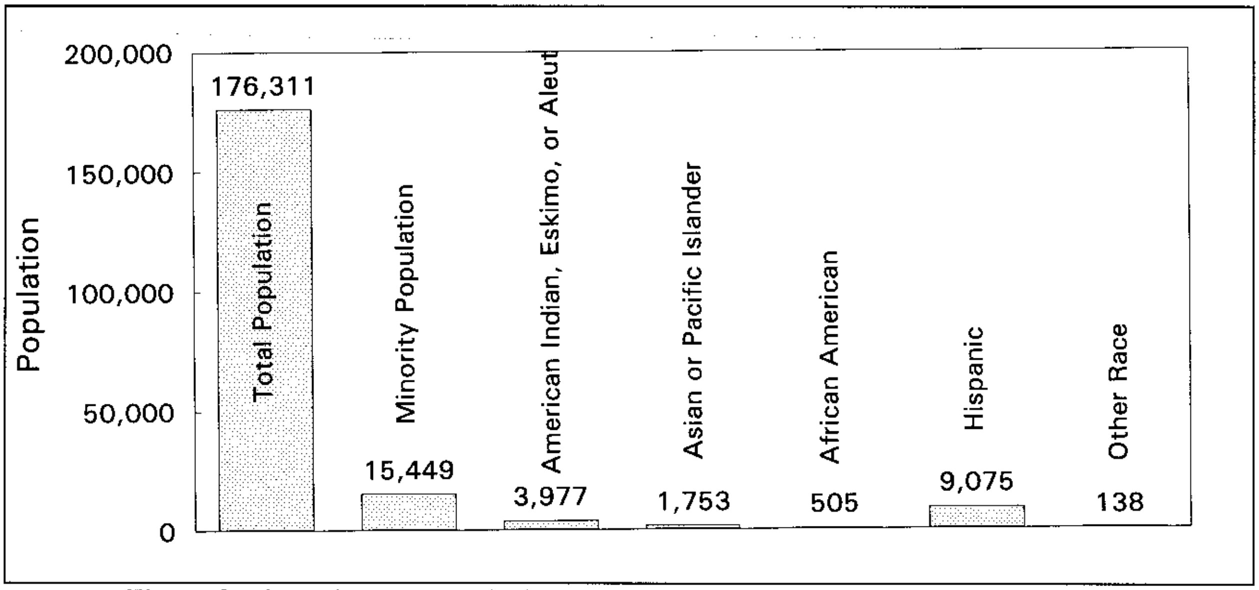


Figure 3-56 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Idaho National Engineering Laboratory

Seventeen public school districts provide educational services for 56,899 school-aged children within the seven-county region. Most of the public schools in the seven-county region operate at levels at or above the design capacity of their classroom facilities.

There are 18 fire districts in the seven-county area surrounding the Idaho National Engineering Laboratory. These 18 districts operate a total of 30 fire stations staffed by 179 paid and 313 volunteer firefighters. Bingham, Bonneville, Butte, Clark, and Jefferson counties, which surround the Idaho National Engineering Laboratory, have developed emergency plans to be implemented in the event of a radiological or hazardous materials emergency. Eight hospitals serve the seven-county region. The Eastern Idaho Regional Medical Center in Idaho Falls, with 311 beds, is the largest hospital in the region. Occupancy rates range from 22.0 to 89.2 percent in the region. Law enforcement services in the

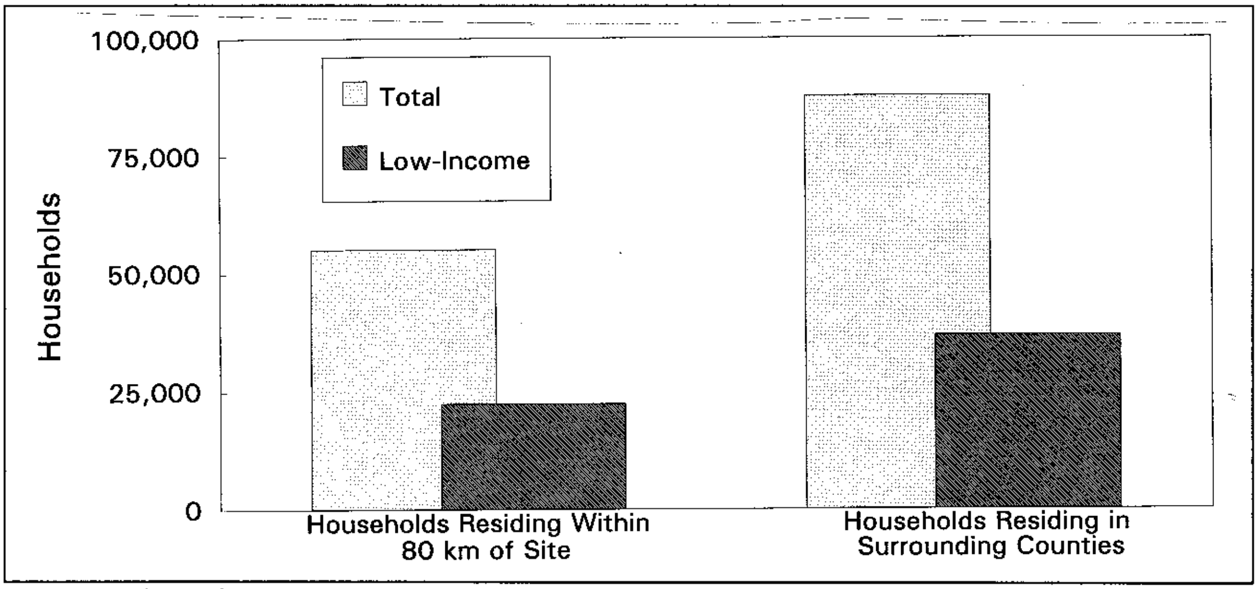


Figure 3-57 Low-Income Households Residing within 80 km (50 mi) of the Idaho National Engineering Laboratory

seven-county area are provided by sheriff's offices in each county, 12 city police departments, and the Idaho State Police. Clark County has the highest ratio of law enforcement personnel per 1,000 people (6.3 per 1,000), and Butte County has the lowest ratio (1.3 per 1,000).

3.3.2.10 Historical, Archaeological, and Cultural Resources

In the course of more than 100 cultural resource surveys, 1,506 cultural resources have been discovered and recorded within the Idaho National Engineering Laboratory boundaries (DOE, 1995c), not including architectural properties directly associated with the creation and operation of the Idaho National Engineering Laboratory. Only four percent of the Idaho National Engineering Laboratory has been surveyed, however, and most surveys were conducted near major facility areas. The 1,506 resources recorded at the Idaho National Engineering Laboratory through 1992 include 688 prehistoric sites, 38 historic sites, 753 prehistoric isolated finds, and 27 historic isolated finds (DOE, 1995c; Gilbert and Ringe, 1993). Until formal evaluations can be performed, all of the cultural sites are considered to be potentially eligible for the National Register of Historic Places. Only Goodale's Cutoff, part of the Northern Alternate of the Oregon Trail, is listed on the National Register of Historic Places. All of the isolated find artifacts have been categorized as unlikely to meet eligibility requirements (Yohe, 1993).

Most of the 688 prehistoric archaeological sites located to date at the Idaho National Engineering Laboratory are classified as lithic scatters. These sites consist of more than 10 stone artifacts, but lack evidence of other categories of cultural materials or features. At 12 percent of these sites, limited subsurface excavations have been conducted. Prehistoric cultural resources vary in density and type across the Idaho National Engineering Laboratory, but appear to be associated with definable physical features of the land. Areas classified as having very high sensitivity have been identified along the Big Lost River, atop buttes, and within craters and caves. The southernmost part of the Lemhi Mountains, the Lake Terreton basin, and a 2,800 m (9,200 ft) wide zone along the edge of lava fields are classified as high-sensitivity areas. With the exception of the Central Facilities Area and Experimental Breeder Reactor-I, which are located in high-sensitivity zones, all developed Idaho National Engineering Laboratory facility areas are located in or on the edge of low or medium sensitivity areas.

As of March, 1993, 38 historic sites and 27 historic isolated finds had been recorded on the Idaho National Engineering Laboratory (Gilbert and Ringe, 1993). These resources include small homesteads and irrigation canals, temporary campsites of sheep and cattle drivers, stage and wagon roads, and some mining-related features.

In most instances, a property must be at least 50 years old to be considered for inclusion in the National Register of Historic Places, unless it meets certain criteria. The unique nature of existing DOE site facilities at the Idaho National Engineering Laboratory and their historic role in nuclear research and development make them eligible for special consideration regardless of their age. Two buildings dating from 1942 appear on a partial inventory of potentially significant historic the Idaho National Engineering Laboratory facilities (Braun et al., 1993). The Experimental Breeder Reactor-I is listed in the National Register of Historic Places, and is designated as a National Historic Landmark. Although the Experimental Breeder Reactor-I is the only nuclear property less than 50 years old that has been formally listed in the National Register, other DOE facilities eligible for listing have been identified, including the Auxiliary Reactor Areas I, II, III; Boiling Water Reactor Experiment V; Materials Test Reactor; Engineering Test Reactor; and the Test Area North hangar (Braun et al., 1993). The Idaho State Historic Preservation Office believes that most major structures related to nuclear research at the Idaho National Engineering Laboratory are probably eligible for the National Register (Yohe, 1993).

As of July 1993, several specific types of locations have been identified as associated with traditional religious practices. The most significant of these locations are the East and Middle Buttes located in the southern section of the Idaho National Engineering Laboratory, and more than 20 lava tube and blister cave sites, which are considered sacred by the Shoshone-Bannock Tribes. The Tribes also consider archaeological sites to be sensitive resources, especially rock art sites. The entire the Idaho National Engineering Laboratory falls within the traditional territory of the Shoshone-Bannock Tribes. These areas of concern are likely to be located within very high sensitivity zones identified for presence of prehistoric archaeological resources, such as the Big Lost River, Birch Creek, buttes, craters, caves, lava edge zones, Lemhi Mountains, and the Lake Terreton basin

3.3.3 Description of the Affected Environment at the Hanford Site

This section describes the potentially affected environment of the Hanford Site. The location of the Hanford Site is shown in Figure 3-58.

3.3.3.1 Geology

The region of the Pacific Northwest that contains the Hanford Site lies within the Columbia Intermontane physiographic province, which is bordered on the north and east by the Rocky Mountains, and on the west by the Cascade Range (Figure 3-58). The province has been a topographic and structural depression since the early Miocene period, and is subdivided into smaller physiographic units. The dominant geologic characteristics of this province are the thick accumulations of basaltic lava flows extending laterally from central Washington eastward into Idaho, and southward into Oregon (Tallman et al., 1979). The ancient basalt surface has been subsequently modified by tectonism, volcanism, weathering, and erosion.

The Columbia Intermontane Province is divided into four subprovinces. The Hanford Site is contained within the Columbia Basin subprovince, which contains most of the Columbia River Basalt Group. The Columbia Basin subprovince is further divided into six physiographic sections, with the Hanford Site located in parts of the Yakima Folds and the Central Plains sections. Much of the Columbia Basin subprovince was affected by preglacial cataclysmic flooding associated with the sudden release of water from glacial Lake Missoula. Cataclysmic floods were responsible for much of the present morphology of

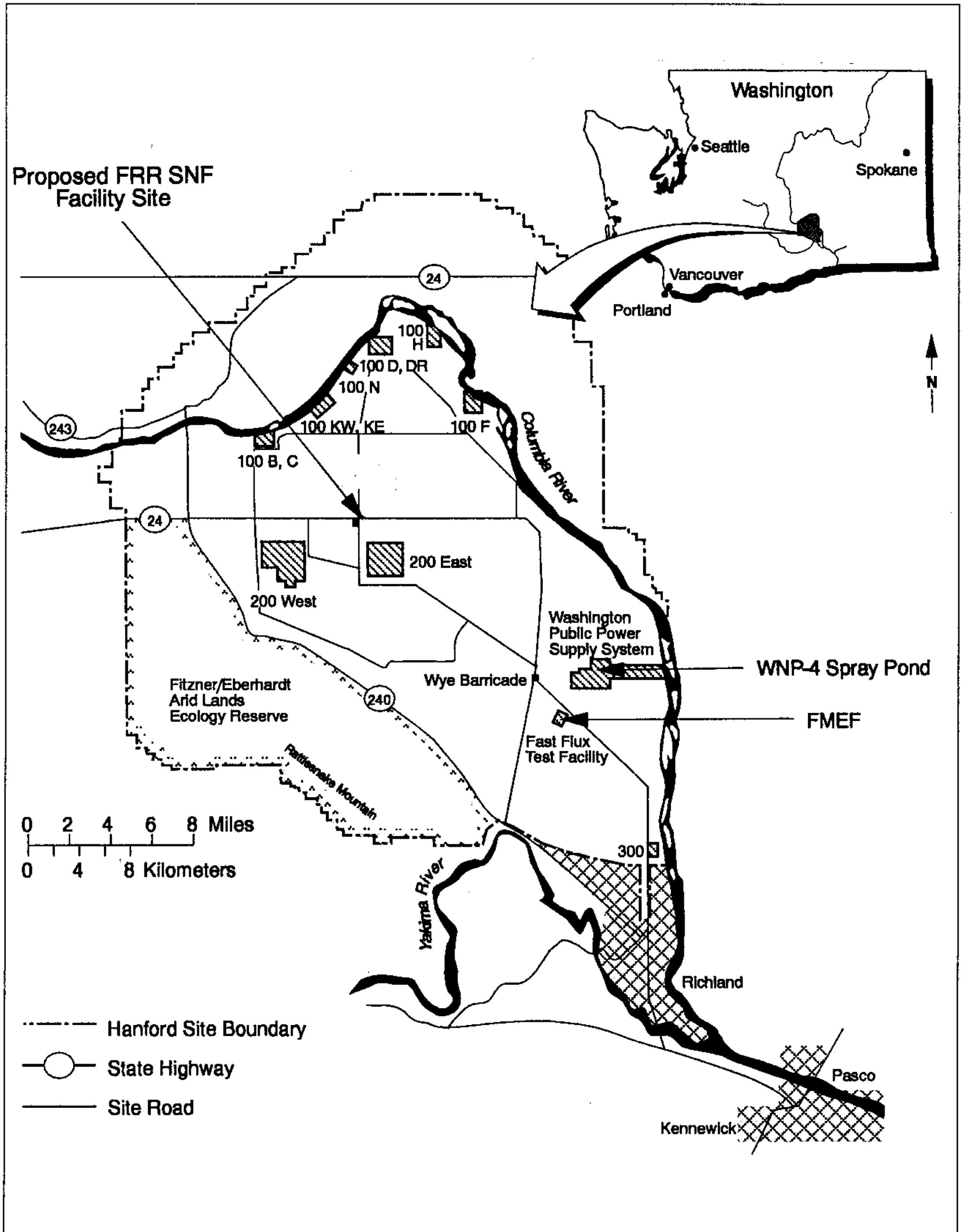


Figure 3-58 Location of the Hanford Site

the Channeled Scabland and Central Plains section. Fluvial and lacustrine processes associated with the ancestral Columbia River system have been active since the late Miocene. Sedimentary deposits indicate that deposition was continuous from about 10.5 million years ago until about 3.5 million years ago (DOE, 1988c). Quaternary volcanism has been limited to the extreme western margin of the Columbia Basin subprovince, and is associated with the Cascade Range Province.

The Hanford Site is located within the Pasco Basin, which was formed by the deformation of the lava flows into broad structural and topographic basins. The Pasco Basin is defined by anticlinal structures of basaltic rock known as the Saddle Mountains to the north, the Umtanum Ridge, Yakima Ridge, and Rattlesnake Hills to the west, and a series of doubly plunging anticlines merging with the Horse Heaven Hills to the south.

Most known faults within the region are associated with anticlinal fold axes, and were probably formed concurrently with the folding (DOE, 1986a). Existing known faults within the Hanford Site area include tear faults with lengths of up to 3 km (1.9 mi) on Gable Mountain, and the Rattlesnake-Wallula alignment. Strike-slip faults have not been observed crosscutting the Pasco Basin. Structures within the Hanford Site have shown the greatest deformation along the hinge area of the anticlinal ridges, decreasing with distance from that area (i.e., the greatest amount of tectonic jointing and faulting occurs in the hinge zone and decreases toward the gently dipping limbs). The faults usually exhibit low dips with small displacements, may be confined to the layer in which they occur, and die out to no recognizable displacement in short lateral distances (DOE, 1986a).

Fifteen different soil types present on the Hanford Site have been listed and described (Hajek, 1966). The soil types vary from sand to silty and sandy loam.

3.3.3.2 Seismology and Volcanology

Seismicity of the Columbia Plateau, as determined by the rate of earthquakes per area and the historical magnitude of these events, is relatively low compared to other regions of the Pacific Northwest, the Puget Sound area, and western Montana/eastern Idaho. Figure 3-59 shows the locations of all earthquakes that occurred in the Columbia Plateau from 1850 to 1969 with Modified Mercalli Intensity of 5 or greater. Swarms of small, shallow earthquakes lasting from several weeks to months, and clustered in an area 5 to 10 km (3 to 6 mi) in lateral dimension are the predominant seismic events. Earthquake swarms may contain from four to more than 100 earthquakes of magnitude 1.0 to 3.5. Detailed locations of swarm earthquakes indicate that the events occur on fault planes of variable orientation and not on a single throughgoing fault plane (DOE, 1995c).

Shallow earthquake swarm activity in the central Columbia Plateau is concentrated principally north and east of the Hanford Site. Here, earthquakes of magnitude greater than 3.0 occur, with the largest recorded (in 1973) having magnitude 4.4 north of the Hanford Site. Deeper earthquakes occur in the central Columbia Plateau, although at much lower frequencies than the shallower swarm events. Deep seismic activity generally occurs randomly, and is not associated with known geologic structures or with patterns of shallow seismicity. Earthquake focal mechanisms in the central Columbia Plateau generally indicate reverse faulting on east-west planes, consistent with a north-south directed maximum compressive stress, and with the formation of the east-west oriented anticlinal fold of the Yakima Fold Belt (Rohay, 1987). The earthquake focal mechanisms in the western margin of the Columbia Plateau also indicate north-south compression, but here the minimum compressive stress is oriented east-west, resulting in strike-slip faulting (Rohay, 1987).

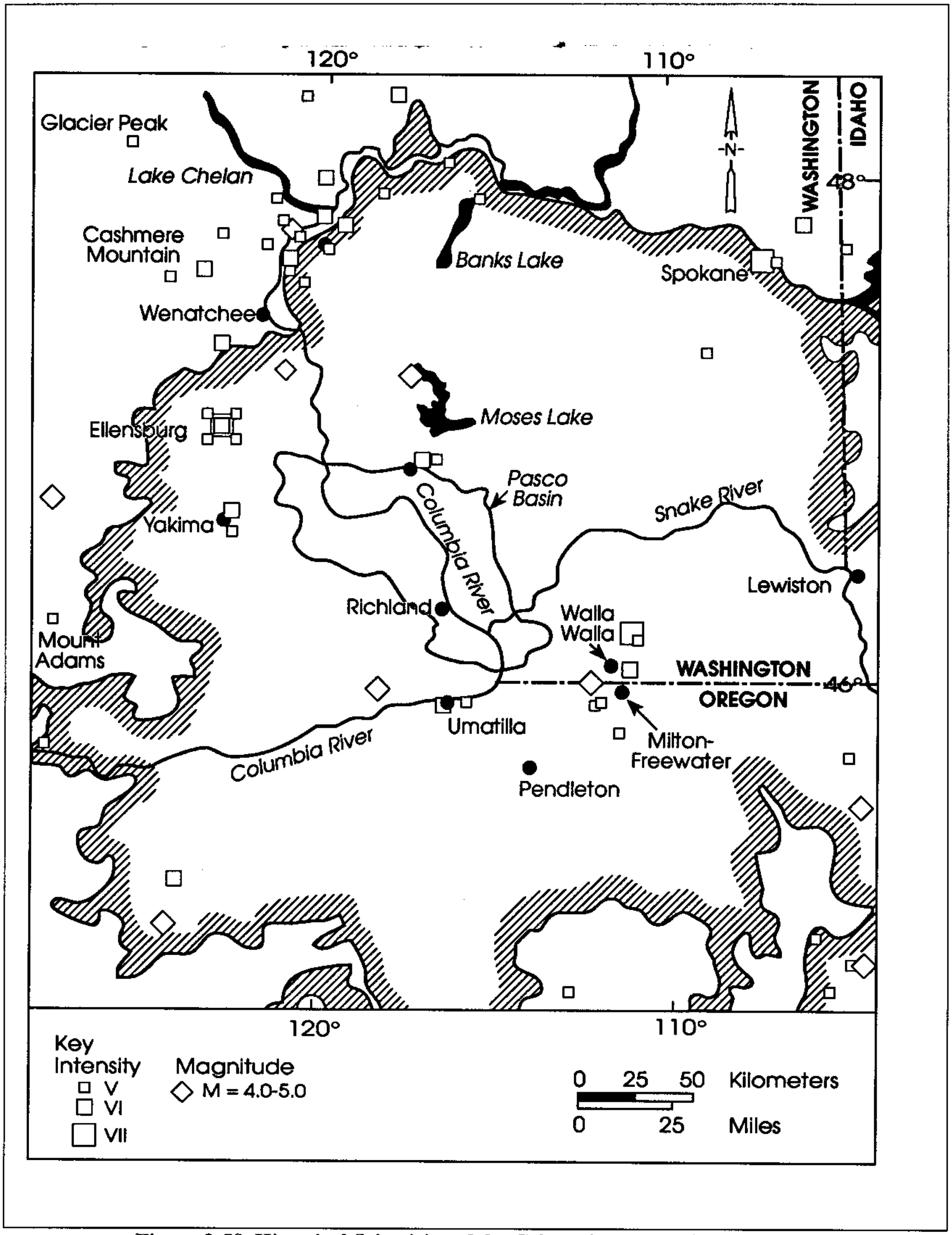


Figure 3-59 Historical Seismicity of the Columbia Plateau (DOE, 1995c)

There are several major volcanoes in the Cascade Range west of the Hanford Site. The nearest volcano is Mount Adams, about 165 km (103 mi) from the Hanford Site, and the most active is Mount St. Helens, approximately 220 km (137 mi) west-southwest from the Hanford Site.

3.3.3.3 Hydrology

The major geologic units of the Hanford Site are, in ascending order: basement rocks of unknown origin and composition, the Columbia River Basalt Group with interbedded sediments of the Ellensburg Formation, the Ringold Formation, the Plio-Pleistocene unit, and the Hanford formation.

3.3.3.3.1 Surface Water

The Hanford Site occupies approximately 33 percent of the land area within the Pasco Basin. Primary surface-water features associated with the Hanford Site are the Columbia and Yakima Rivers. Several surface ponds and ditches are present, and are generally associated with fuel and waste processing activities (Figure 3-60). There is no significant surface water flow from the Hanford Site to nearby rivers.

Cold Creek and its tributary, Dry Creek, are ephemeral streams within the Yakima River drainage system along the southern boundary of the Hanford Site. Rattlesnake Springs, located on the western part of the Hanford Site, forms a small surface stream that flows for about 3 km (1.9 mi) before disappearing into the ground.

Normal river elevations within the Hanford Site range from 120 m (396 ft) above mean sea level when the river enters the Hanford Site near Vernita, to 104 m (343 ft) where it leaves the Hanford Site near the 300 Area.

Large Columbia River floods have occurred in the past (DOE, 1986a), but the likelihood of recurrence of large-scale flooding has been reduced by the construction of several flood control/water storage dams upstream from the Hanford Site. The maximum historical flood on record occurred June 7, 1894, with a peak discharge at the Hanford Site of 21,000 m^3 per sec (741,600 ft^3 per sec) (Figure 3-61). The largest recent flood took place in 1948, with an observed peak discharge of 20,000 m^3 per sec (706,300 ft^3 per sec) at the Hanford Site. The probability of flooding at the magnitude of the 1894 and 1948 floods has been greatly lowered because of upstream regulation by dams (Figure 3-60). There have been fewer than 20 major floods on the Yakima River since 1862 (DOE, 1986a).

The probable maximum flood for the Columbia River below Priest Rapids Dam has been calculated to be 40,000 m^3 per sec (1,412,600 ft^3 per sec). This flood would inundate the 100 Areas located adjacent to the Columbia River, but the central portion of the Hanford Site would remain unaffected (DOE, 1986a).

The U.S. Army Corps of Engineers evaluated a number of scenarios on the effects of failures of Grand Coulee Dam, assuming flow conditions on the order of 11,000 m^3 per sec (388,500 ft^3 per sec). The discharge resulting from a 50 percent breach at the outfall of Grand Coulee Dam was determined to be 600,000 m^3 per sec (21,188,800 ft^3 per sec). In addition to the areas inundated by the probable maximum flood, the remainder of the 100 Areas, the 300 Area, and nearly all of Richland, WA, would be flooded (DOE, 1986a).

Surface Water Quality: The Washington State Department of Ecology classifies the Columbia River as Class A (excellent) between Grand Coulee Dam and the mouth of the river near Astoria, OR (DOE, 1986a). The Class A designation requires that industrial uses of this water be compatible with other uses, including drinking water, wildlife, and recreation.

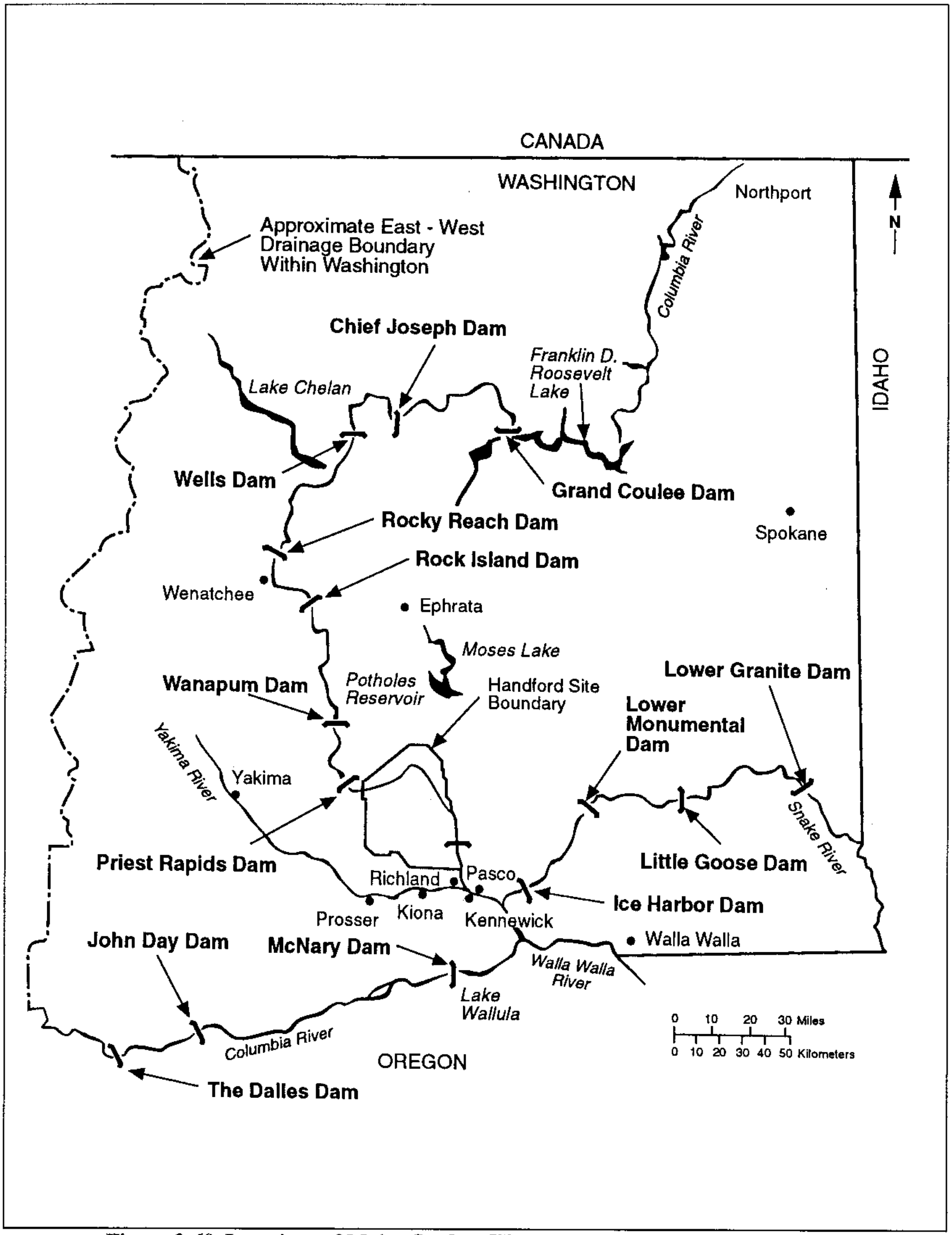


Figure 3-60 Locations of Major Surface Water Resources and Principal Dams within the Columbia Plateau

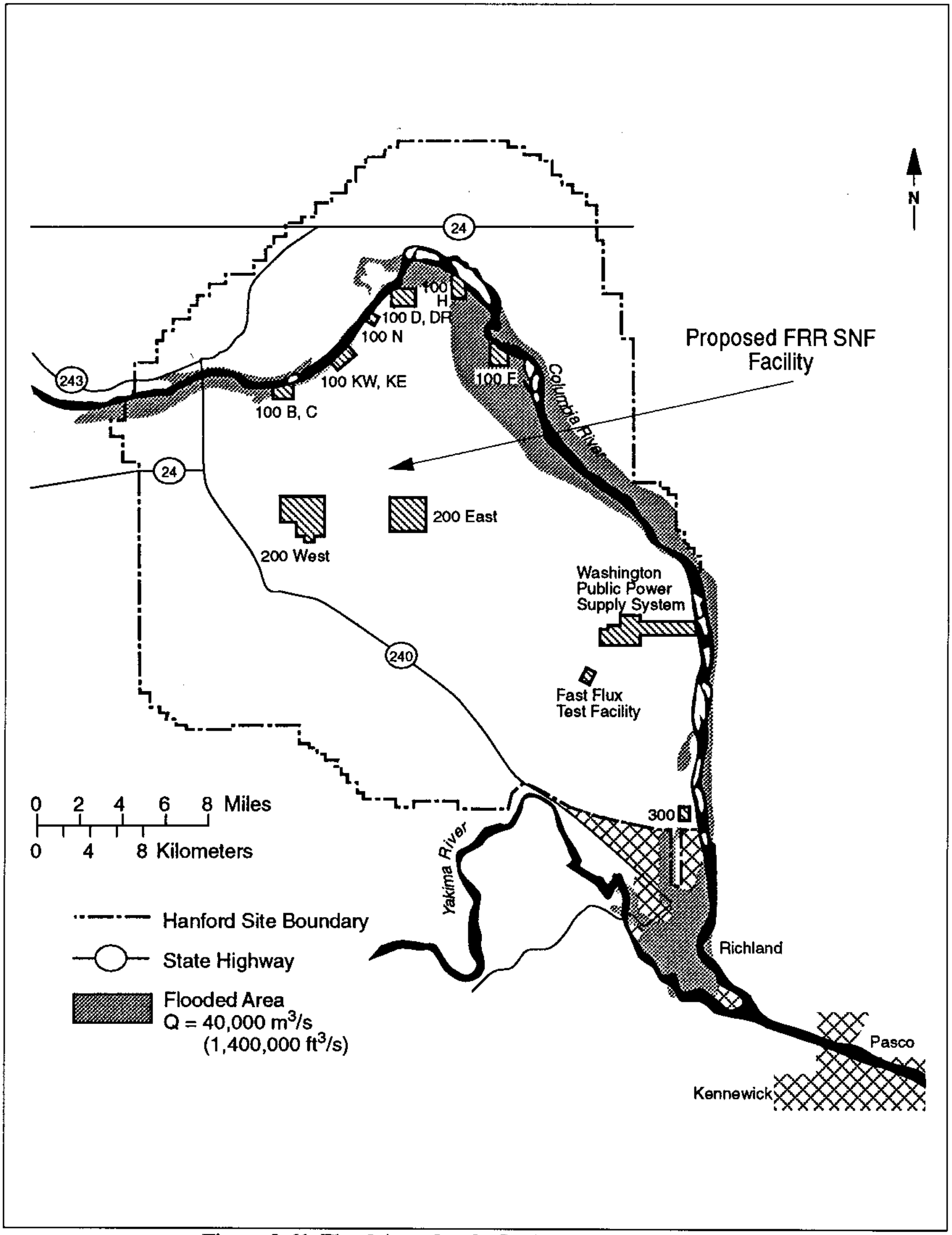


Figure 3-61 Flood Area for the Probable Maximum Flood

Radiological monitoring shows low levels of radionuclides in samples of Columbia River water. Hydrogen-3 (tritium), iodine-129, and uranium are found in slightly higher concentrations downstream from the Hanford Site than upstream (DOE, 1995c), but were well below concentration guidelines established by DOE and U.S. Environmental Protection Agency drinking water standards. Cobalt-60 and iodine-131 were not consistently found in measurable quantities during 1987 samples of Columbia River water from Priest Rapids Dam, the 300-Area water intake, or the Richland City pumphouse. The average annual strontium-90 concentrations were essentially the same at Priest Rapids Dam and the Richland Pumphouse for 1987, and were well below the State of Washington and U.S. Environmental Protection Agency drinking water standards.

3.3.3.3.2 Groundwater

Groundwater under the Hanford Site occurs in unconfined and confined conditions. The unconfined aquifer is contained within the glaciofluvial sands and gravel and within the Ringold Formation. The bottom of the aquifer is the basalt surface or, in some areas, the clay zones of the lower member of the Ringold Formation. The confined aquifers consist of sedimentary interbeds and/or interflow zones that occur between dense basalt flows in the Columbia River Basalt Group. The main water-bearing portions of the interflow zones occur within a network of interconnecting vesicles and fractures of the flow tops or flow bottoms.

Sources of natural recharge to the unconfined aquifer are rainfall and runoff from the higher bordering elevations, water infiltrating from small ephemeral streams, and river water along influent reaches of the Yakima River and Columbia River.

From the recharge areas to the west, the groundwater flows downgradient to the discharge areas, primarily along the Columbia River. This general west-to-east flow pattern is interrupted locally by the groundwater mounds in the 200 Areas. From the 200 Areas, there is also a component of groundwater flow to the north between Gable Mountain and Gable Butte. These flow directions represent current conditions; the aquifer is dynamic and responds to changes in natural and artificial recharge.

Local recharge to the shallow basalts is believed to result from infiltration of precipitation and runoff along the margins of the Pasco Basin. Regional recharge of the deep basalts is thought to result from interbasin groundwater movement originating northeast and northwest of the Pasco Basin in areas where the Wanapum and Grande Ronde Basalts crop out extensively. Groundwater discharge from the shallow basalt is probably to the overlying unconfined aquifer and the Columbia River. The discharge area(s) for the deep groundwater is currently uncertain, but flow is believed to be generally southeastward with discharge speculated to be south of the Hanford Site (DOE, 1986a).

Groundwater Quality: The groundwater composition is that of a dilute (less than or approximately 350 mg per L total dissolved solids) calcium-bicarbonate chemical type. Other principal chemical constituents include sulfate, silica, magnesium, and nitrate (the latter contributed through the disposal of chemical reprocessing waters).

Contamination of the groundwater is caused by releases from various liquid-waste disposal facilities. Nitrate and tritium contamination has migrated away from these sites in a general west-to-east direction. Some longer-lived radionuclides such as strontium-90 and cesium-137 have reached the groundwater, primarily through liquid-waste disposal cribs. Small quantities of longer-lived radionuclides have reached the water table via a failed groundwater monitoring well casing and through reverse well injection, a disposal practice which was discontinued at the Hanford Site in 1947 (DOE, 1995c).

Areal and stratigraphic changes in groundwater chemistry characterize basalt groundwaters beneath the Hanford Site (Graham et al., 1981). The stratigraphic position of these changes is believed to delineate flow system boundaries, and to identify chemical evolution taking place along groundwater flow paths. Overall, waters of the shallow basalts are of a sodium-bicarbonate chemical type, while those of the deep basalts are of a sodium-chloride chemical type (DOE, 1986a). Iodine-129 and tritium have been detected in confined groundwater in the Saddle Mountains Basalt beneath the Hanford Site (DOE, 1986a).

3.3.3.4 Meteorology

The Hanford Site is located in a semi-arid region of southeastern Washington State. A summary of the following data (through 1980) has been published (Stone et al., 1983).

Wind: Prevailing wind directions on the 200-Area Plateau are from the northwest in all months of the year. Monthly and annual joint frequency distributions of wind direction versus wind speed for the Hanford Meteorological Station have been recorded (Stone et al., 1983). Monthly average wind speeds are lowest during the winter months, averaging 10 to 11 km per hr (6.2 to 6.8 mph), and highest during the summer, averaging 14 to 16 km per hr (8.7 to 9.9 mph). High winds are also associated with thunderstorms. The average occurrence of thunderstorms is 10 per year. Although thunderstorms are most frequent during the summer, they have occurred in each month (DOE, 1995c). It is estimated that the probability of a tornado striking a point at the Hanford Site is 0.0000096 per year, or less than one in one-hundred thousand.

Temperature and Humidity: Diurnal and monthly averages and extremes of temperature, dew point, and humidity have been recorded (Stone et al., 1983). Ranges of daily maximum and minimum temperatures vary from normal maxima of 2°C (35.6°F) in early January to 35°C (95°F) in late July. The record maximum temperature is 46°C (114.8F), and the record minimum temperature is -32.8°C (-27.0°F). The annual average relative humidity at the Hanford Meteorological Station is 54 percent. It is highest during the winter months, averaging about 75 percent, and lowest during the summer, averaging about 35 percent.

Precipitation: Average annual precipitation at the Hanford Meteorological Station is 16 cm (6.3 in). Most of the precipitation occurs during the winter, with nearly half of the annual amount occurring in the months of November through February. Winter monthly average snowfall ranges from 0.8 cm (0.3 in) in March to 13.5 cm (5.3 in) in January. Snowfall accounts for about 38 percent of all precipitation during the months of December through February.

Atmospheric Dispersion: Good dispersion conditions associated with neutral and unstable stratification exist about 57 percent of the time during the summer.

Air Quality: Air quality in the vicinity of the Hanford Site is generally classified as quite good. Wind-eroded dust, resulting from plowed fields and arid terrain with sparse vegetation, is an occasional problem in the area. The atmospheric conditions that produce the dust are favorable to pollutant transport and diffusion.

3.3.3.5 Ecology

The Hanford Site is made up of relatively large, undisturbed (1,450 km^2, about 560 mi^2) expanses of shrub-steppe desert that contain numerous plant and animal species suited to the region's semi-arid environment. The Hanford Site consists of mostly undeveloped land, with facilities only occupying about 6 percent of the total available area. Most of the Hanford Site has not experienced tillage or livestock grazing since the early 1940's.

The Hanford Site vegetation has been characterized as a shrub-steppe (DOE, 1995c) with relatively low productivity. In the early 1800s, the dominant plant was the big sagebrush with an understory of perennial bunchgrasses, but with the advent of livestock and crop raising, the natural vegetation was overtaken by what is today the dominant plant, cheatgrass. Some planted trees exist and serve as nesting platforms for several species of birds, including hawks, owls, ravens, magpies, and great blue herons, and as night roosts for wintering bald eagles (Rickard and Watson, 1985). Today, the vegetation picture at the Hanford Site consists of eight major kinds of plant communities: sagebrush/bluebunch wheatgrass, sagebrush/cheatgrass or sagebrush/Sandberg's bluegrass, sagebrush-bitterbrush/cheatgrass, greasewood/cheatgrass-saltgrass, winterfat/Sandberg's bluegrass, thyme buckwheat/Sandberg's bluegrass, cheatgrass-tumble mustard, willow or riparian, spiny hopsage, and sand dunes. More than 600 species of plants have been identified at the Hanford Site (Sackschewsky et al., 1992). A distribution of the dominant plant communities is shown in Figure 3-62.

More than 300 species of terrestrial and aquatic insects have been found on the Hanford Site. Grasshoppers and darkling beetles are among the more conspicuous groups, and along with other species, are important in the food web of the local birds and mammals. Approximately 12 species of amphibians and reptiles are known to exist on the Hanford Site (Fitzner and Gray, 1991). The most abundant reptile is the side-blotched lizard, although short-horned and sagebrush lizards are also seen frequently. Also common are gopher snakes, yellow-bellied racers, the Pacific rattlesnake, toads, and frogs.

More than 125 species of birds have been identified on the Hanford Site (Rogers and Rickard, 1977). The horned lark and western meadowlark are the most abundant nesting birds in the shrub-steppe. The Hanford Site supports populations of chukar partridge, gray partridge, and sage grouse, with the greatest concentration in the Rattlesnake Hills. Wastewater ponds at the Hanford Site are important habitats for songbirds, shore birds, ducks, and geese. The most important waterfowl is the Canada goose. Hawks and owls use the Hanford Site as a refuge, especially during nesting (DOE, 1995c).

Approximately 39 species of mammals have been identified on the Hanford Site (Fitzner and Gray, 1991). Most are small and nocturnal. Of this group, the Great Basin pocket mouse is the most abundant, and others include the deer mouse, Townsend ground squirrel, pocket gopher, harvest mouse, Norway rat, sagebrush vole, grasshopper mouse, vagrant, Leasts chipmunk, and Merriam vole. Larger mammals include the mule deer and elk. The largest vertebrate predator inhabiting the Hanford Site is the coyote. A herd of wild, free-roaming elk is centered almost entirely on the Arid Lands Ecology Reserve, a part of the Hanford Site established as an environmental research study area in 1968.

The Columbia River is the dominant aquatic ecosystem, and the river supports a large and diverse community of plankton, benthic invertebrates, fish, and other communities. Plankton populations in the Hanford reach are influenced by communities that develop in the reservoir of upstream dams, with phytoplankton and zooplankton populations being largely transient, flowing from one reservoir to another. Phytoplankton species include diatoms, golden or yellow-brown algae, red algae, and dinoflagellates. Zooplankton populations in the Hanford reach of the Columbia are generally sparse. All major freshwater benthic taxa are represented in the river. Forty-three species of fish had been identified in the river (DOE, 1995c), and since then the brown bullhead has been collected, bringing the number to 44. Of these, the Chinook salmon, sockeye salmon, coho salmon, and steelhead trout use the river as a migration route to and from upstream spawning areas, and are of the greatest economic importance. Small spring streams contain diverse biotic communities and are extremely productive, consisting of dense blooms of watercress and aquatic insects. Temporary wastewater ponds and ditches on the Hanford Site develop riparian communities and become quite attractive to migrating birds in autumn and spring.

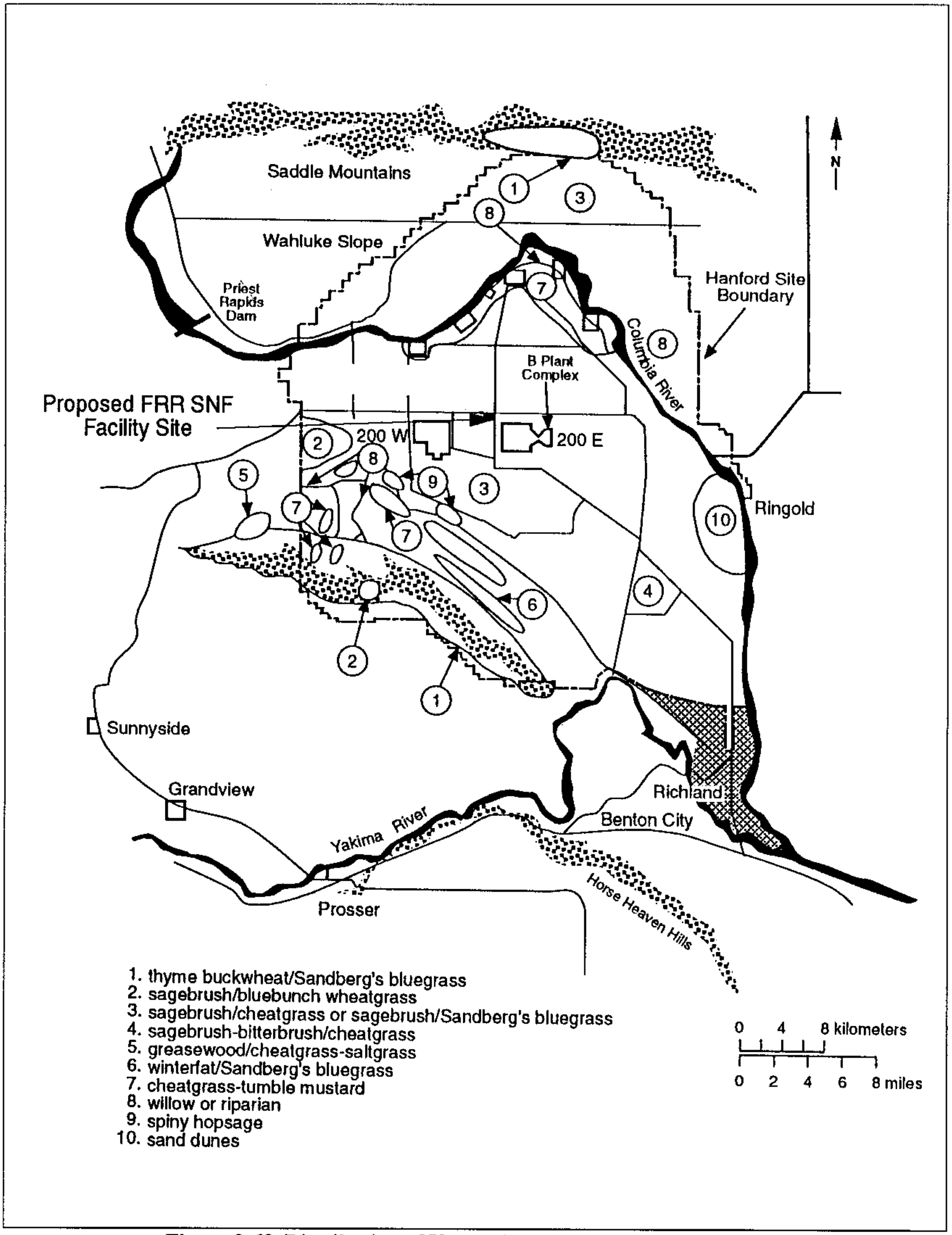


Figure 3-62 Distribution of Vegetation Types on the Hanford Site

Threatened, Endangered, and Candidate Plant and Animal Species: Threatened and endangered species, as listed by both the Federal Government and the State of Washington, are shown in Table 4.9-1 of Appendix A, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). No plants or mammals on the Federal list are known to occur on the site. There are, however, several species of plants and animals that are under consideration for formal listing. Two species of plants are included in the State listing. Columbia milk-vetch, found on dry land benches along the river, is listed as threatened, and yellowcress, found on the wetted zone of the water's edge, is designated as endangered. The Federal Government lists the American peregrine falcon as endangered and the bald eagle as threatened. The State of Washington lists (in addition to the peregrine falcon and bald eagle) the white pelican and sandhill crane as endangered, and the ferruginous hawk as threatened. The peregrine falcon is a casual migrant to the Hanford Site and does not nest there. The bald eagle is a regular winter resident, foraging on dead salmon and waterfowl along the river, but not nesting on the Hanford Site. Ferruginous hawks have increasingly used power poles for nesting sites. Mammals considered endangered by the State are the pygmy rabbit, the Merriam shrew, the pallid bat, and the long-eared myotis.

Several small spring streams, wetlands, temporary water bodies, and national and State wildlife refuges are located on, or adjacent to, the Hanford Site.

3.3.3.6 Land Use

The Hanford Site encompasses 1,450 km^2 (560 mi^2), and includes several DOE operational areas. The major areas are:

- The entire Hanford Site, which has been designated a National Environmental Research Park;
- The 100 Areas, bordering on the right bank (south shore) of the Columbia River, which are the sites of the eight retired plutonium production reactors. The 100 Areas occupy about 11 km^2 (4.2 mi^2);
- The 200-West and 200-East Areas are located on a plateau about 8 and 11 km (5.0 and 6.8 mi), respectively, from the Columbia River. For some time, these areas have been dedicated to fuel reprocessing and waste processing management and disposal activities. The 200 Areas cover about 16 km^2 (6.2 mi^2);
- The 300 Area, located just north of the city of Richland, is the site of nuclear research and development and nuclear fuel fabrication. This area covers 1.5 km^2 (0.6 mi^2);
- The 400 Area covers about 0.6 km^2 (0.25mi^2) and is about 8 km (5 mi) north of the 300 Area and is the site of the Fast Flux Test Facility used in the testing of breeder reactor systems. Also included in this area is the Fuels and Material Examination Facility;
- The 600 Area includes all of the Hanford Site not occupied by the 100, 200, 300, or 400 Areas. Land uses within the 600 Area include:
  - The Arid Lands Ecology Reserve, a 310 km^2 (120 mi^2) tract set aside for ecological studies,
  - 4 km^2 (1.5 mi^2) leased by the State of Washington, part of which is used for low-level radioactive waste disposal,

- 4.4 km^2 (1.7 mi^2) for Washington Public Power Supply System nuclear power plants,
- 2.6 km^2 (1.0 mi^2) transferred to the State of Washington as a potential site for the disposal of nonradioactive hazardous wastes,
- About 130 km^2 (50 mi^2) under revocable use permit to U.S. Fish and Wildlife Refuge,
- 225 km^2 (87 mi^2) under revocable use permit to the Washington State Department of Game for recreational game management,
- Support facilities for the controlled access areas, and
- Laser Intercrometer Gravitational Wave Observatory.

Surrounding the Hanford Site, 660 km^2 (255 mi^2) have been designated for Arid Lands Ecology Reserve, U.S. Fish and Wildlife Refuge, and Washington State Department of Game (DOE, 1986a).

Land use in other areas includes urban and industrial development, irrigated and dry-land farming, and grazing. In 1985, wheat represented the largest single crop in terms of area planted in Benton and Franklin counties with 116,145 ha (287,000 acres). Corn, alfalfa, hay, and barley are other major crops in Benton and Franklin counties.

In 1986, the Columbia Basin Project, a major irrigation project to the north of the Tri-Cities (Richland, Pasco, and Kennewick), produced gross crop returns of $343 million, representing 19 percent of all crops grown in Washington State. In 1986, the average gross crop value per irrigated ha was $1,640 ($664 per irrigated acre). The largest percentages of irrigated acres produced alfalfa hay (29.4 percent of irrigated acres), wheat (15.0 percent), and corn (feed grain) (9.4 percent). Other significant crops are potatoes, apples, dry beans, asparagus, and pea seed.

3.3.3.7 Noise

Studies at the Hanford Site dealing with the propagation of noise have dealt primarily with occupational noise at work sites. Environmental noise levels have not been extensively evaluated because of the remoteness of most the Hanford activities, and isolation from receptors that are covered by Federal or State statutes. This discussion will focus on what little environmental noise data is available. The majority of available information consists of model predictions, which in many cases have not been verified, since these predictions indicate that the potential to violate Federal or State standards is remote or unrealistic.

The Noise Control Act of 1972 and its subsequent amendments (Quiet Communities Act of 1978, 42 USC 4901-4918, 40 CFR 201-211) directs the regulation of environmental noise to the State. The State of Washington has adopted RCW 70.107, which authorizes the Department of Ecology to implement rules consistent with Federal noise control legislation. RCW 70.107 and the implementing regulations embodied in WAC 173-60 through 173-70 define the regulation of environmental noise levels. Maximum noise levels are defined for the zoning of the area for the environmental designation for noise abatement. The Hanford Site is classified as a Class C environmental designation for noise abatement area on the basis of industrial activities. Unoccupied areas are also classified as Class C areas by default, because they are neither Class A (residential) nor Class B (commercial). Maximum noise levels are established based on the environmental designation for noise abatement classification of the receiving area and the source area (DOE, 1995c).

3.3.3.8 Transportation

The Tri-Cities serve as a regional transportation and distribution center with major air, land, and river connections. The Tri-Cities area has direct rail service, provided by Burlington Northern and Union Pacific, which connects the area to more than 35 States (Figure 3-63). Union Pacific operates the largest fleet of refrigerated railcars in the United States, and is essential to food processors that ship frozen food from this area. Passenger rail service is provided by Amtrak, which has a station in Pasco.

The Hanford Site infrequently uses docking facilities at the ports of Benton on the Columbia River. No barge accidents were reported in 1988 (DOE, 1995c).

Daily air passenger and freight services connect the area with most major cities through the Tri-Cities Airport in Pasco. The airport is currently served by two commuter-regional and two national airlines. The main runway is 2,350 m (7,755 ft) in length, and can accommodate landings and takeoffs by medium-range commercial aircraft, such as the Boeing 727-200 and Douglas DC-9. Two additional airports, located in Richland and Kennewick, are limited to serving private aircraft.

The Tri-Cities are linked to the region by five major roads. Route 395 joins the area with Spokane to the northeast. Both route 395 and route 240, which crosses through the Hanford Site, connect with Interstate 90 to the north. Route 12 links the region with Yakima to the northwest, Lewiston, ID to the east, and Walla Walla to the southeast. Finally, the area is linked to Interstate 84 to the south, via Interstate 82 and Route 14. Routes 240 and 24 traverse the Hanford Site and are maintained by the State of Washington. Other roads within the Hanford Site are maintained by DOE.

3.3.3.9 Socioeconomics

The level of operations at the Hanford Site plays a dominant role in the socioeconomics of the Tri-Cities and other parts of Benton and Franklin counties. The agricultural community also has a significant effect on the local economy. Any major changes in the Hanford Site operations will affect the Tri-Cities and other areas of Benton and Franklin counties.

Three major sectors have been the principal driving forces of the economy in the Tri-Cities since the early 1970's: (1) DOE and its contractors operating the Hanford Site, (2) Washington Public Power Supply System in its construction and operation of nuclear power plants, and (3) the agricultural community, including a substantial food processing component. Most of the goods and services produced by these sectors are exported outside the Tri-Cities. In addition to direct employment and payrolls, these major sectors also support a sizable number of jobs in the local economy through their procurement of equipment, supplies, and business services. Three other components can be identified a contributors to the economic base of the Tri-Cities economy. These include other employers, such as: (1) Siemens Nuclear Power Corporation in North Richland, (2) Sandvik Special Metals in Kennewick, and (3) Boise-Cascade's Wallula corrugated paper mill, tourism, and Government transfer payments in the form of pension benefits.

The Hanford Site dominates the local employment picture with more than one-quarter of the total nonagricultural jobs in Benton and Franklin counties (15,552 out of 64,300), so the Hanford Site payroll impacts the Tri-Cities and State economy. In 1991, the Hanford Site employment accounted directly for

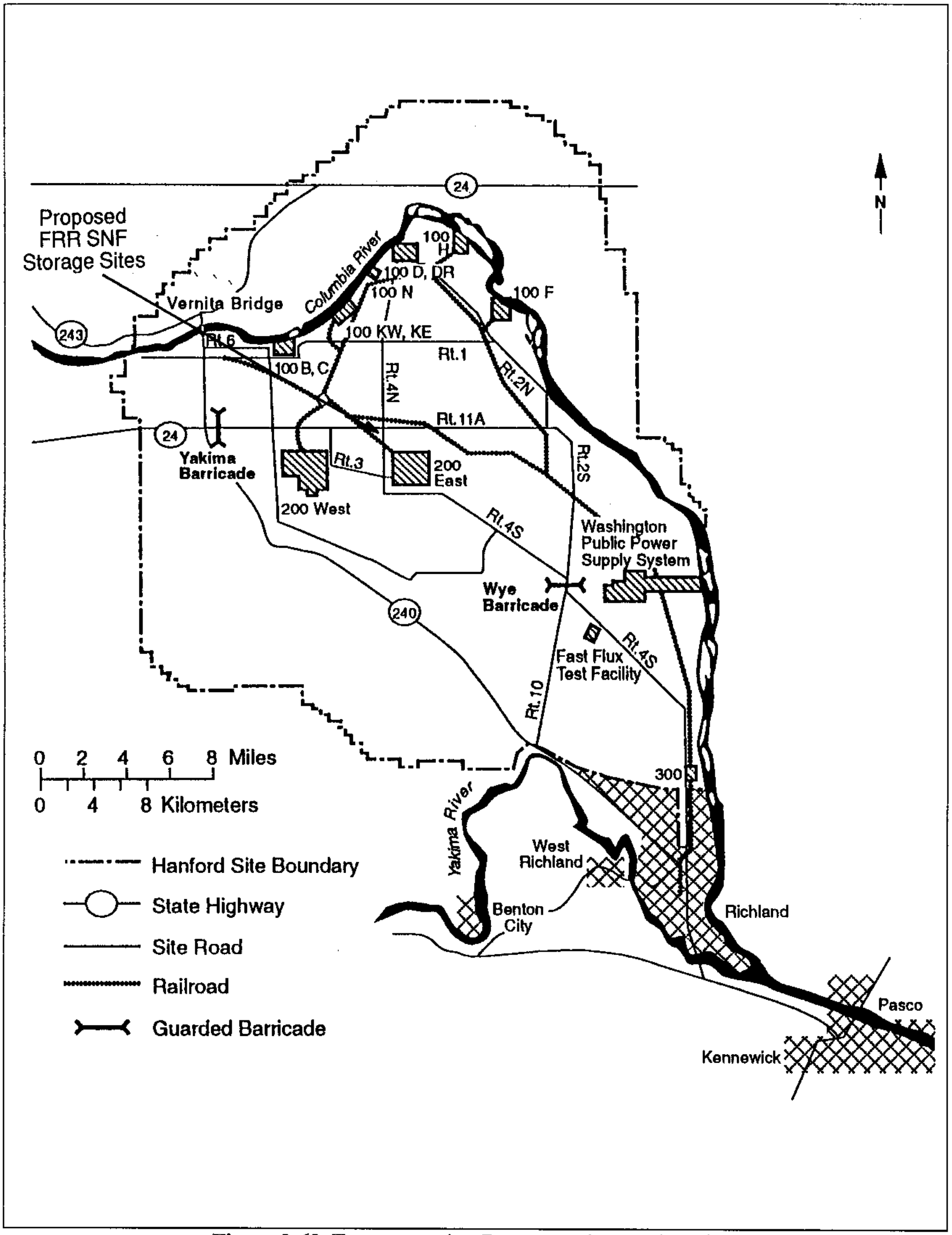


Figure 3-63 Transportation Routes on the Hanford Site

24 percent of total nonagricultural employment in Benton and Franklin counties, and slightly more than 0.6 percent of all nonagricultural Statewide jobs. In 1991, the Hanford Site operations directly accounted for an estimated 42 percent of the payroll dollars earned in the area.

The Washington Public Power Supply System continues to be a major employer in Richland, with more than 1,700 workers and an approximate $71.6 million in payroll during the year. In 1990, agriculture was responsible for nearly 12,900 jobs, nearly 17 percent of total employment. This includes about 2,200 farm proprietors accounting for $121 million in crop and livestock production which provides 7,600 wage and salary jobs, and "agri-business" (farm and ranch supporting activities such as application of fertilizers, sales of farm supplies and equipment, etc.) accounting for 900 jobs. Employment in the food processing sector included 20 food processors producing potato products, canned fruits and vegetables, wine, and animal feed.

Other major employers include about 3,500 workers in Benton and Franklin counties. Tourism has increased in the area, and overall tourism expenditures in the Tri-Cities were roughly $77.5 million in 1990. In 1990, 15,903 people over the age of 65 resided in Benton and Franklin counties. This segment of the population supports the local economy on the basis of income received from Government transfer payments and pensions, private pension benefits, and prior individual savings. A summary of estimated major Government pension benefits received by the residents of Benton and Franklin counties in 1990 is shown in Table 4.3-7 of Appendix A, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The estimated income of this component of the basic sector is roughly equivalent to the entire agricultural sector.

Estimates by the U.S. Bureau of the Census for 1990 place Benton and Franklin counties' population totals at 112,560 and 37,473, respectively. Within each county, the 1990 estimates distribute the Tri-Cities population as follows: Richland, 32,315; Kennewick, 42,159; and Pasco, 20,337. The 1990 estimates of racial categories by the Bureau of the Census indicate that in Benton and Franklin counties, Asians represent a lower proportion, and individuals of Hispanic origin represent a higher proportion of the racial distribution than those in the State of Washington. Fifty-six percent of the population of Benton and Franklin counties is under the age of 35, compared to 54 percent for the State of Washington, and the 65-yr-old and older group constitutes 10 percent of the population, compared to 12 percent for the State.

Social and economic impacts of the Hanford Site operations are concentrated in Benton County, Franklin County, and the Tri-Cities area made up of Pasco, Richland, and Kennewick. The region of influence for the Hanford Site is represented by the 80 km (50 mi) radius around the site. Figure 3-64 represents the general ethnic characteristics of the population within the 80 km (50 mi) radius. Low income data for the region of influence is shown in Figure 3-65. A low-income household is one with an income of 80 percent or lower than the median income of the county. Approximately 42 percent of the households in the region of influence are low income families.

In 1990, nearly 92 percent of all housing (38,781 total units) in the Tri-Cities was occupied. Single-unit housing, which represents nearly 58 percent of the total units, has a 96 percent occupancy rate. Multiple-unit housing has an occupancy rate of nearly 91 percent. Pasco has the lowest occupancy rate, 89 percent in all categories of housing, followed by Kennewick (93 percent) and Richland (94 percent). Representing nine percent of the housing unit types, mobile homes have an 81 percent occupancy rate.

Primary and secondary education are served by the Richland, Kennewick, Pasco, and Kiona-Benton school districts. In 1992, spring enrollment for all districts was approximately 24,876 students. Post-secondary education in the Tri-Cities area is provided by a junior college, Columbia Basin College, and the Tri-Cities branch campus of Washington State University.

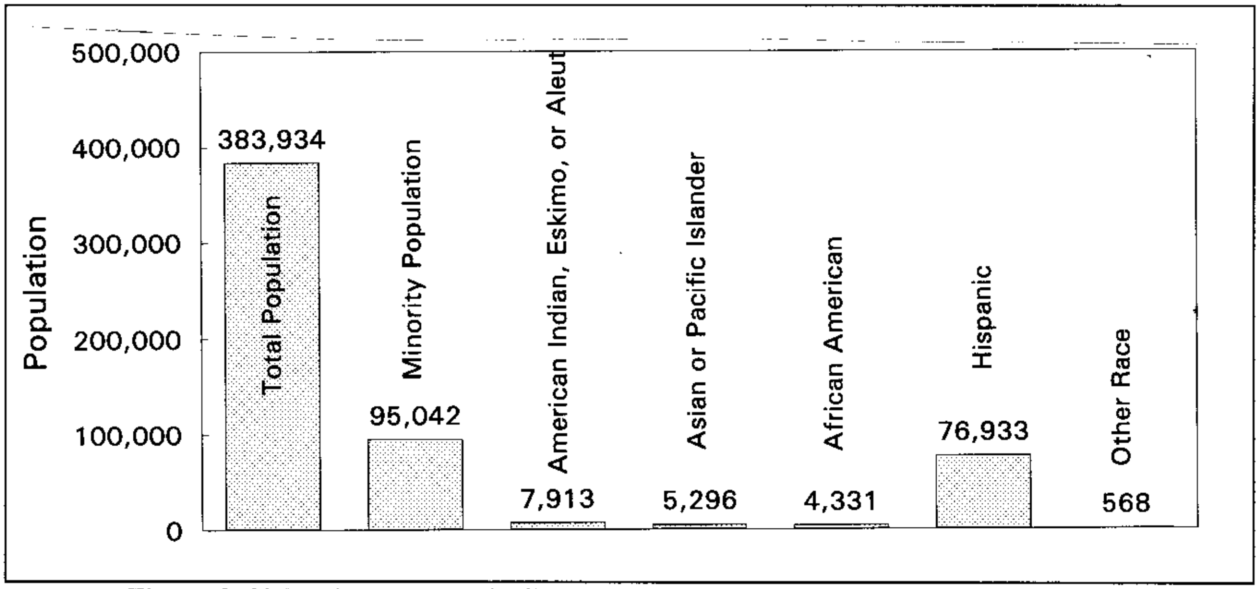


Figure 3-64 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Hanford Site

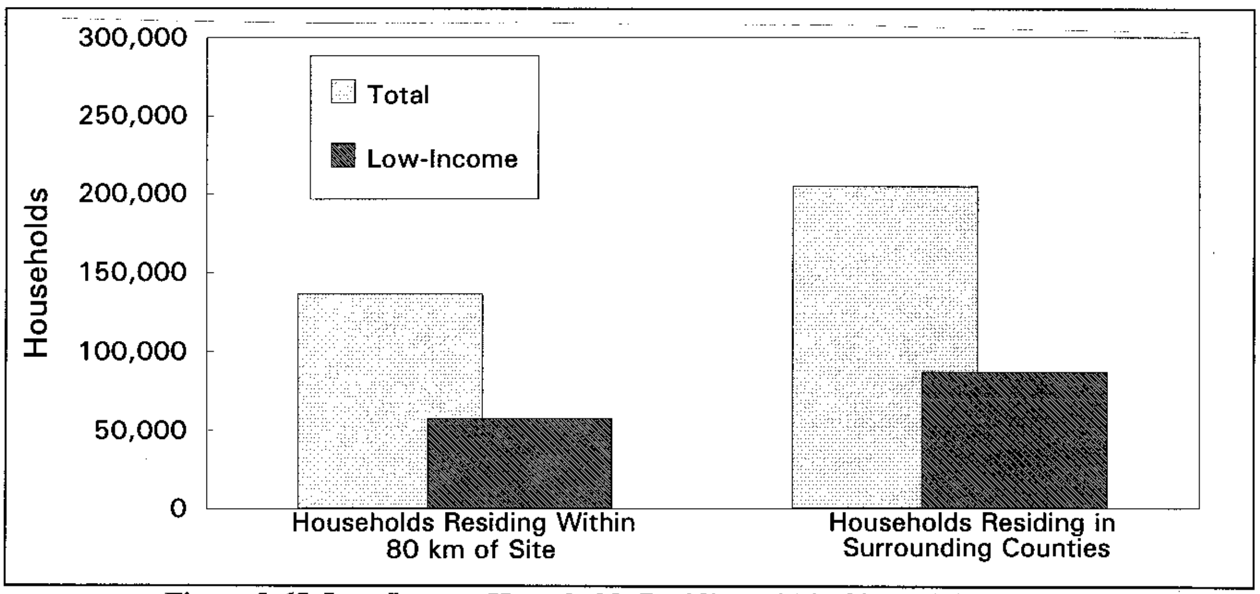


Figure 3-65 Low-Income Households Residing within 80 km (50 mi) of the Hanford Site

The Tri-Cities have three major hospitals and four minor emergency centers. Kadlec Medical Center, located in Richland, has 136 beds and functions at an average of 39.5 percent capacity. Kadlec Medical Center's 5,754 annual admissions represent more than 42 percent of the Tri-Cities market. Kennewick General Hospital maintains an average 45.5 percent occupancy rate of its 71 beds with 3,619 admissions. Our Lady of Lourdes Hospital, located in Pasco, reports an average occupancy rate of 36.5 percent.

Police and fire protection in Benton and Franklin counties is provided by Benton and Franklin county sheriff's departments, local municipal police departments, and the Washington State Patrol Division, headquartered in Kennewick. The Hanford Fire Patrol is composed of 126 trained firefighters.

The principal source of water in the Tri-Cities and the Hanford Site is the Columbia River, from which the water systems of Richland, Pasco, and Kennewick draw a large portion of the average 43 billion liters (11.4 billion gal) used in 1991. Electricity is provided by the Benton County Public Utility District, Benton Rural Electric Association, Franklin County Public Utility District, and city of Richland Energy Services Department. In the Pacific Northwest, hydropower, and to a lesser extent, coal and nuclear power, constitute the region's electrical generation system. Throughout the 1980's, the Northwest had more electric power than it required, and was operating at a surplus. This surplus has been exhausted, and there is only approximately enough power supplied by the existing system to meet the current electricity needs.

3.3.3.10 Historical, Archaeological, and Cultural Resources

The Hanford Site is rich in cultural resources, and contains numerous well-preserved archaeological sites representing both prehistoric and historical periods. The area is considered a homeland by many Native Americans.

More than 10,000 years of prehistoric human activity in the Middle Columbia River region have left extensive archaeological deposits along river shores (DOE, 1995c) and well-watered areas inland from the river (DOE, 1995c; Greene, 1975;). Graves are common in various settings, and spirit quest monuments are found on high, rocky summits (Rice, 1968). By virtue of their inclusion in the Hanford Site, from which the public is restricted, archaeological deposits found in the Hanford reach of the Columbia River, on adjacent plateaus, and mountains have been spared some of the disturbances that have befallen other sites.

There are currently 248 prehistoric archaeological sites recorded in the files of the Washington State Office of Archaeology and Historic Preservation. Forty-seven of these sites are included on the National Register of Historic Places, two as single sites (45BN121, the Hanford Island Site; 45GR137, Paris Site), and the rest in seven archaeological districts. In addition, a nomination has been prepared for one cultural district (Gable Mountain/Gable Butte), and a renomination for two additional archaeological districts is pending (Wahluke, Coyote Rapids). Two other sites, 45BN90 and 45BN412, are considered eligible for the National Register. Archaeological sites include remains of numerous pithouse villages, various types of open campsites, and cemeteries along the riverbanks (DOE, 1995c), spirit quest monuments, hunting camps, game drive complexes and quarries in mountains and rocky bluffs, hunting/kill sites in lowland stabilized dunes, and small temporary camps near perennial sources of water located away from the river (Rice, 1968).

Native American people of various tribal affiliations have populated the Hanford reach of the Columbia River since prehistoric and early historic times. Wanapums and Yakama people of the Chamnapum band, and some of their descendants still live nearby, while others have been incorporated into the Yakama and Umatilla Reservations. Palus people, who lived on the lower Snake River, joined the Wanapum and Chamnapum to fish the Hanford reach and inhabited the river's west bank (DOE, 1995c). Walla Walla and Umatilla people also made periodic visits to the area to fish. These groups retain traditional and secular ties to the region, with native plant and animal foods, some found on the Hanford Site, being used in ceremonies. Certain landmarks, especially Rattlesnake Mountain, Gable Mountain, Gable Butte, Goose Egg Hill and others along the river are considered sacred. The many cemeteries found along the river are considered sacred by these groups.

Two hundred-two historic archaeological sites and 11 other historic localities have been recorded in the published literature. Localities include the Allard Pumping Plant at Coyote Rapids, the Hanford Irrigation Ditch, the Hanford townsite, Wahluke Ferry, the White Bluffs townsite, the Richmond Ferry, Arrowsmith townsite, a cabin at East White Bluffs ferry landing, the White Bluffs road, the old Hanford high school, and the Cobblestone Warehouse at Riverland (DOE, 1995c). In addition to recorded sites, numerous data from additional unrecorded sites, including homesteads, corrals, and dumps, have been recorded by the Hanford Cultural Resources Laboratory since 1987. The 100-B Reactor has been listed on the National Register of Historic Places. Other Manhattan Project facilities remain to be evaluated.

3.3.4 Description of the Affected Environment at the Oak Ridge Reservation

The Oak Ridge Reservation is a key DOE site hosting three separate facilities with missions including basic and applied research and development; storage of special nuclear materials; weapons dismantlement, storage, and evaluation; and environmental restoration and waste management. The site is operated for DOE by Martin Marietta Energy Systems. This section describes the potentially affected environment at the Oak Ridge Reservation.

3.3.4.1 Geology

The Oak Ridge Reservation lies within the western portion of the Valley and Ridge Province, near the boundary with the Cumberland Plateau, in the State of Tennessee (Figure 3-66). The Valley and Ridge Province is characterized by numerous linear ridges and valleys that trend approximately southwest-northeast. A generalized geologic map of the Oak Ridge Reservation is shown in Figure 3-67. The Oak Ridge Reservation is mostly underlain by the Rome Formation and Conasauga, Knox, and Chickamauga Groups, sedimentary rocks of Cambrian and Ordovician age. A detailed description of these formations is given in the Programmatic SNF&INEL Final EIS (DOE, 1995c).

Sinkholes, large springs, caves, and other karst features are common in the Knox Group, and those parts of the Oak Ridge Reservation underlain by limestones and dolomites are classified as karst terrains. Although no site-specific geologic characterization has been conducted at the West Bear Creek Valley site, it appears that the proposed site for the construction of a foreign research reactor spent nuclear fuel storage facility is located over the lower Conasauga Group strata not normally characterized by karst development.

The soils found in the Oak Ridge Reservation vicinity generally contain clay minerals, chert, and quartz sand (Hatcher et al., 1992). Soils on the Oak Ridge Reservation tend to retain moisture, and are typically 90 percent saturated below a depth of 3 m (10 ft) (Ketelle and Huff, 1984). Depth of soil profiles on the Oak Ridge Reservation vary from 15 cm (5.9 in) on slopes, to 18 m (60 ft) over dolomites in the Knox Group (Boyle et al., 1982).

3.3.4.2 Seismology and Volcanology

The Oak Ridge Reservation is located in a region of moderate seismic activity, having an average of one to two earthquakes per year, with seismic activity occurring in bursts followed by long periods of inactivity. From 1811 to 1975, only five major earthquakes or earthquake series have affected the Oak Ridge Reservation area. No deformation of recent surface deposits has been detected, and seismic shocks from the surrounding, more seismically active areas, are dissipated by distance from the epicenters (Boyle et al., 1982). During the 1811 to 1975 period, none of the earthquakes were of a magnitude that caused severe damage to buildings or structures.

The underlying structure of the Oak Ridge Reservation is complex due to the extensive faulting and deformation characteristic of the region. There are three regional thrust faults in the Oak Ridge Reservation area, the Kingston, Whiteoak Mountain, and Copper Creek Faults. All three strike to the

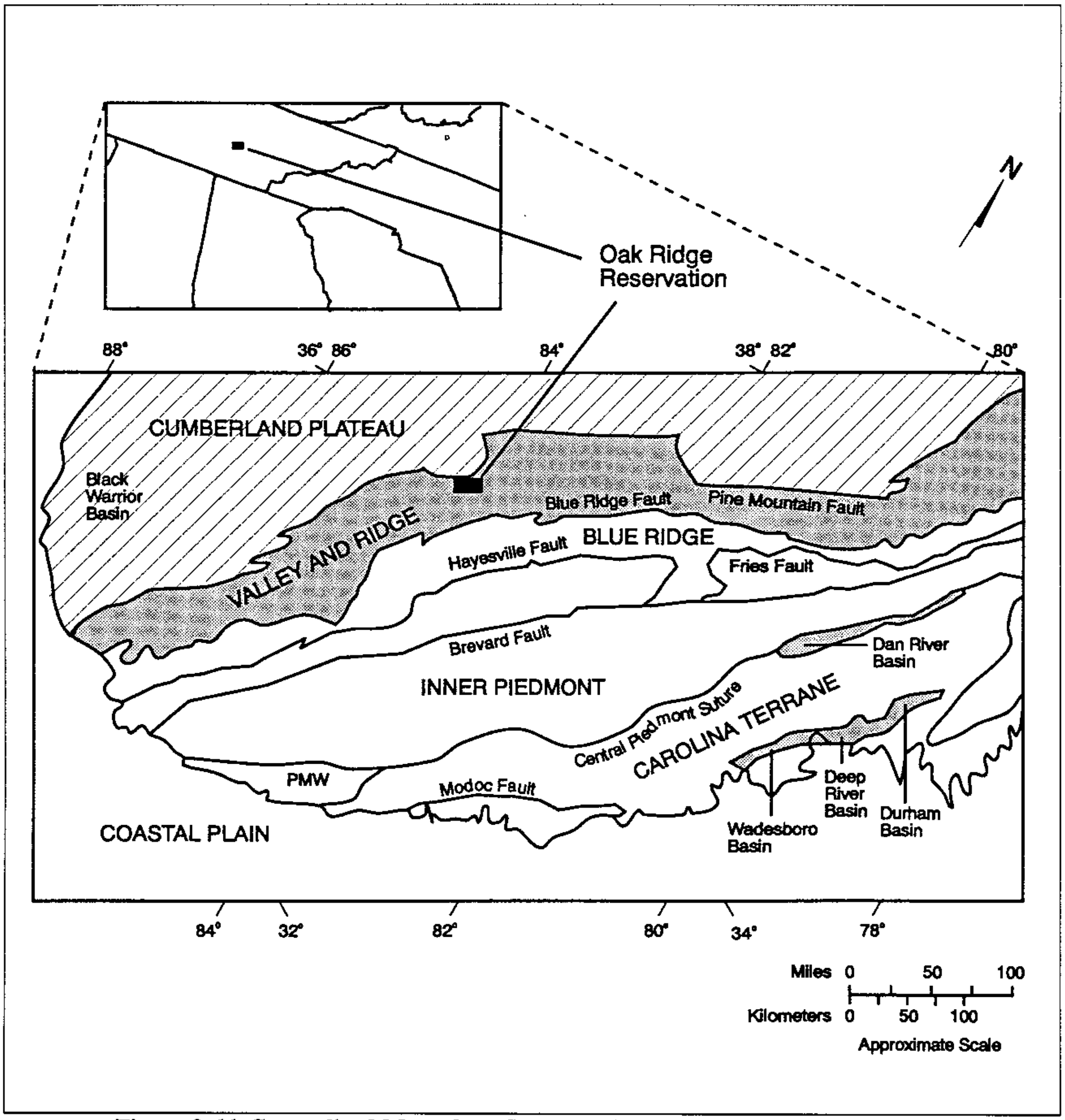


Figure 3-66 Generalized Map of the Southern Appalachian Geologic Provinces Showing the Location of the Oak Ridge Reservation

northeast and dip to the southeast. The most recent movement on the faults was during the Late Pennsylvanian/Early Permian periods (280 to 290 million years ago), and consequently, the faults are not considered to be capable at present (Butz, 1984).

There is no evidence that there has been significant volcanic activity in the vicinity of the Oak Ridge Reservation for more than 1 million years (DOE, 1995c). Three studies conducted specifically for the Oak Ridge Reservation have been summarized (Beavers et al., 1982).

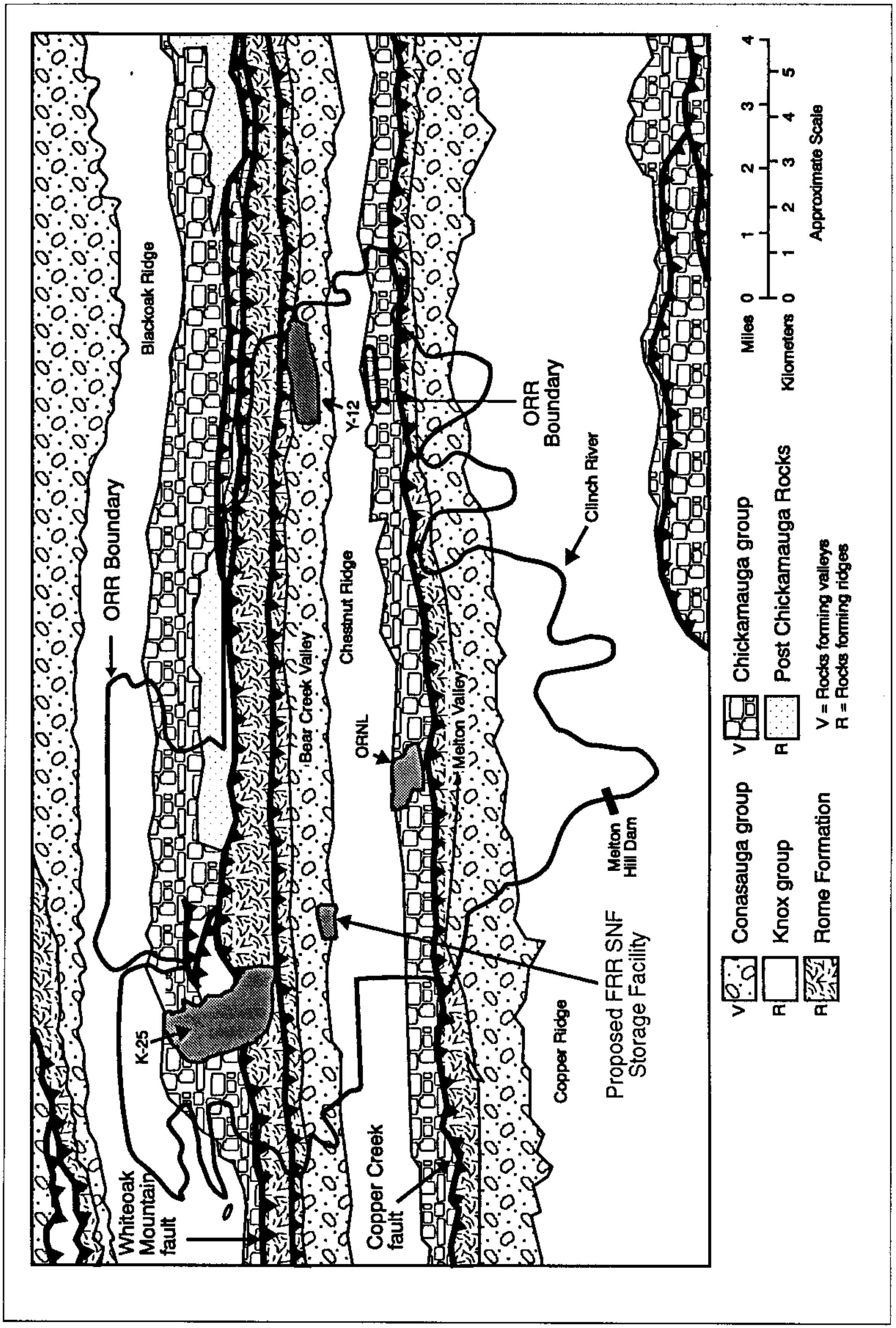


Figure 3-67 Geologic Map of the Oak Ridge Reservation

3.3.4.3 Hydrology

3.3.4.3.1 Surface Water

The hydrologic system on the Oak Ridge Reservation is controlled by the Clinch River (MMES, 1993a). Its drainage area is about 11,422 km^2 (4,410 mi^2) (Boyle et al., 1982). All water that drains from the Oak Ridge Reservation enters the Clinch River, and subsequently the Tennessee River. Flow in the Clinch-Tennessee river system is regulated by multi-purpose dams of the Tennessee Valley Authority.

The Oak Ridge Reservation is drained by a network of tributaries of the Clinch River. A Statewide stream classification system based on water quality, water use, and resident aquatic biota designates most streams on the Oak Ridge Reservation for fish and aquatic life, irrigation and livestock watering (MMES, 1993b). The drainage pattern on the Oak Ridge Reservation is a weakly developed "trellis" pattern (Lee and Ketelle, 1987), and the surface drainage patterns are also affected by karst topography.

Heavy precipitation in the area causes localized flooding, primarily in the city of the Oak Ridge (MMES, 1993a) and along the Clinch River. Figure 3-68 shows approximate 500-yr floodplains. A number of wetlands occur on the Oak Ridge Reservation (MMES, 1993b), including characteristic forested wetlands along creeks, wet meadows and marshes associated with streams and seeps, and emergent communities in shallow embayments and ponds.

Surface Water Quality: Water quality in the Clinch River is affected by the Oak Ridge Reservation activities, by contaminants introduced upstream from the Oak Ridge Reservation, and by flow regulation at the Tennessee Valley Authority dams. Stream impoundment has resulted in a rise in water temperatures, sediment retention, and contaminant adsorption.

The Clinch River supplies most of the water to the Oak Ridge Reservation, the city of Oak Ridge, and other cities along the river (MMES, 1993a). Major water uses in the Oak Ridge area include withdrawals for industrial and public water supplies, commercial and recreational navigation, and other recreational activities such as fishing, boating and swimming. Water for the city of Oak Ridge is withdrawn upstream from any of the Oak Ridge Reservation discharge points. Five public water supplies, including the cities of Kingston and Harriman, TN, are located downstream of the Oak Ridge Reservation (MMES 1993a). These are located 4 km (2.5 mi) above and 34 km (21 mi) below the mouth of Poplar Creek, respectively.

3.3.4.3.2 Groundwater

Groundwater beneath the Oak Ridge Reservation is heavily influenced by the site geologic structure (Solomon et al., 1992). Geologic units of the Oak Ridge Reservation are assigned to two broad hydrologic groups, Knox Aquifer and the Oak Ridge Reservation aquitards. These aquitards may store fairly large volumes of water, but they transmit only limited amounts.

The Knox Aquifer is the only true aquifer of the Oak Ridge Reservation, and is the primary source of sustained natural flow in perennial streams (Solomon et al., 1992). Flow volumes and potential groundwater flow path length in the Knox Aquifer are greater than in the aquitard. No spent nuclear fuel management facilities would be sited in areas overlying the Knox aquifer.

Recently published reports such as "Status Report; A Hydrologic Framework for the Oak Ridge Reservation", and "Status Report on the Geology of the Oak Ridge Reservation" have all used the term "aquitard" to describe the Pumpkin Valley Shale and a number of the other geologic units beneath the Oak Ridge Reservation. No site-specific data are available to determine at what depth Pumpkin Valley Shale is

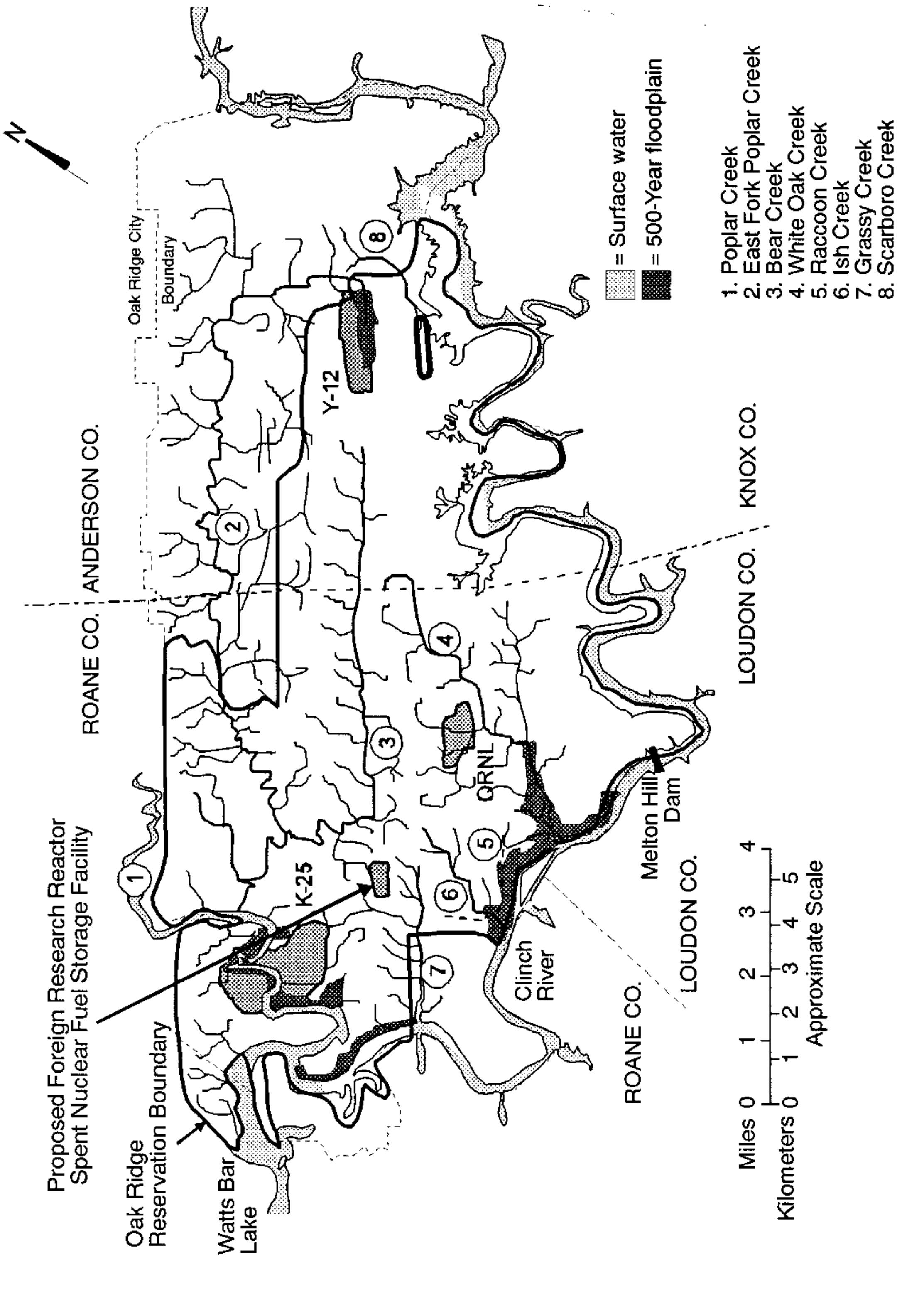


Figure 3-68 Locations of the Clinch River and Tributaries on the Oak Ridge Reservation

encountered at the West Bear Creek Valley, the proposed site for the construction of foreign research reactor spent nuclear fuel storage facilities. It would be logical, however, to think that at depths of 18 m (60 ft) or less on the site, the water-bearing unit most likely to be encountered would be an aquitard (DOE, 1995c).

Groundwater Quality: Background groundwater quality at the Oak Ridge Reservation is generally good in the surficial aquifer zones, and poor in the bedrock aquitards at depths greater than 305 m (1,000 ft) (DOE, 1993d). Groundwater has been contaminated downgradient from waste management facilities and other industrial sources, and discharge of contaminated groundwater has introduced contaminants to streams. Principal groundwater contaminants that exceed applicable standards at the Y-12 Plant include volatile organics, nitrates, heavy metals, and radioactivity (MMES, 1993b). There is one known instance of offsite migration of contaminants from the Oak Ridge Reservation. In 1994, DOE announced that elevated levels of four industrial solvents (carbon tetrachloride, chloroform, tetrachloroethylene, and trichloroethylene) had been detected in the Knox Aquifer monitoring wells east of the Y-12 Plant (Bowdle, 1994). The same solvents are found in groundwater monitoring wells within the Y-12 Plant.

There are no sole-source aquifers beneath the Oak Ridge Reservation (DOE, 1993d). Water rights are not an issue in the region. All groundwater at the Oak Ridge Reservation is classified as Class II (current and potential sources of drinking water and those waters having other beneficial uses).

Although surface water sources provide the main portion of potable water supplies in the area, groundwater does provide for some domestic, municipal, farm, irrigation, and industrial use (MMES, 1993b). Single-family wells are common in areas not served by public water supplies (MMES, 1992). Due to the abundance of surface water and its proximity to the points of use, almost no groundwater is used at the Oak Ridge Reservation (DOE, 1993d).

3.3.4.4 Meteorology

The Oak Ridge Reservation is located within the Great Valley of Tennessee, in which Cumberland Plateau borders to the northwest and the Great Smoky Mountains lie to the southeast. The climate at the Oak Ridge Reservation is influenced by these terrain features.

Wind: The wind direction above the ridgetops and within the valleys tends to follow the orientation of the valleys. The prevailing wind direction is from the southwest, with a secondary maximum from the northeast during the winter, spring, and summer months. This situation is reversed in the fall. Damaging winds are uncommon in the region. Peak gusts recorded in the Great Valley are generally in the 27 to 31 m per sec (60 to 70 mph) range for the months of January through July, in the 22 to 27 m per sec (50 to 60 mph) range for August, September, and December, and in the 16 to 20 m per sec (36 to 45 mph) range in October and November.

Temperature and Humidity: The average daily temperature at the Oak Ridge National Weather Service Station was 14.2°C (57.6°F) for the period of record 1961-1990. The average daily temperature varied from a low of 2.6°C (36.7°F) in January to a high of 24.8°C (76.6°F) in July. The mean relative humidity was 86 percent, with the mean monthly maximum of 92 percent occurring in July and August, and the mean monthly minimum of 80 percent occurring during February and March.

Precipitation: Winter is the wettest of the seasons in the Oak Ridge Reservation area, March and December are the wettest months, and October is the driest. The maximum monthly precipitation was 48.9 cm (19.3 in) in July, while maximum rainfall in a 24-hr period observed at the Oak Ridge National Weather Service was 19 cm (7.5 in), recorded in August 1960 (DOE, 1995c). The annual average precipitation reported by the National Weather Service for the Oak Ridge was 137.2 cm (54 in).

On average, about 51 thunderstorm days per year are recorded at the Oak Ridge National Weather Service station. The Great Valley of Tennessee is infrequently subject to tornadoes. The western half of the State has experienced three times as many tornadoes as the eastern half, where the Oak Ridge Reservation is located. The Oak Ridge Reservation did experience a tornado from a severe thunderstorm on February 21, 1993. The tornado path passed the Y-12 Plant in an east-northeast direction for approximately 21 km (13 mi), ending just north of Knoxville. The wind speeds associated with this tornado ranged from 18 m per sec (40 mph) to nearly 58 m per sec (130 mph), depending on the location along the path (DOE, 1993c). Hurricanes are rarely sustained once they reach as far inland as the Great Valley, due to the rapid loss of energy.

Atmospheric Dispersion: The transport and dispersion of airborne material are direct functions of air movement. The atmospheric conditions are unstable (Stability Classes A through C) approximately 5 percent of the time, neutral (Class D) approximately 43 percent of the time, and stable (Classes E through G) approximately 52 percent of the time at the 10 m (33 ft) level.

Air Quality: A summary of the Oak Ridge Reservation airborne radionuclide emissions for 1992 is presented in Table 4.7-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The maximum effective dose equivalent at the Oak Ridge Reservation boundary is 3.3 mrem as calculated by the GENII code. This is 33 percent of the National Emissions Standards for Hazardous Air Pollutants.

The Oak Ridge Reservation is located in Anderson and Roane Counties in the Eastern Tennessee-Southwestern Virginia Interstate Air Quality Control Region 207. As of 1993, the areas within this Air Quality Control Region were designated as attainment with respect to all National Ambient Air Quality Standards (40 CFR 81.343).

3.3.4.5 Ecology

Land for the Oak Ridge Reservation was primarily in agricultural use, including woodlots and woodlands used for pasture, at the time of acquisition by DOE's predecessor agencies. At least half of the Oak Ridge Reservation was forested. Most of the forest had been partially cut for timber, but not completely cleared on steep slopes. Natural plant communities have re-established themselves on most of the Oak Ridge Reservation, although many areas are maintained as pine plantations or nonforested areas (ORNL, 1988). Approximately 10 percent of the Oak Ridge Reservation has been developed since it was withdrawn from public access, and the remainder of the site has reverted to or been planted with natural vegetation (MMES, 1989).

The vegetation of the Oak Ridge Reservation has been categorized into seven plant communities (Parr and Pounds, 1987). The Oak Hickory forest is one of the most extensive plant communities on the Oak Ridge Reservation. Another abundant plant community is the Pine Hardwood forest and Pine plantations. A total of 899 species, subspecies, and varieties of plants have been identified on the Oak Ridge Reservation (Cunningham and Pounds, 1991). The Oak Ridge Reservation also provides habitat for a large number of animal species. Twenty-six species of amphibians, 33 species of reptiles, 169 species of birds, and 39 species of mammals have been recorded at the Oak Ridge Reservation (Parr and Evans, 1992).

Vegetative communities of the West Bear Creek site are typical of the Oak Ridge Reservation as a whole, composed of second-growth oak-hickory forest and mixed pine-hardwood forest. There are some loblolly pine plantations adjacent to the northern edge of the powerline right-of-way and between the right-of-way and Bear Creek Road (Rosensteel, 1994). Fauna of the site would also be similar to those expected throughout the Oak Ridge Reservation.

Wetlands on the Oak Ridge Reservation include emergent, scrub/shrub, and forested wetland located in embayments of the Melton Hill and Watts Bar Reservoirs that border the Oak Ridge Reservation, along all the major streams, including East Fork Poplar Creek, Poplar Creek, Bear Creek, and their tributaries, in old farm ponds, and around groundwater seeps. Originating on the lower slopes of Pine Ridge are several headwater tributary systems on Grassy Creek that flow from north to south across the West Bear Creek site. The stream valleys contain forested wetlands.

Aquatic habitats on or adjacent to the Oak Ridge Reservation range from small, free-flowing streams in undisturbed watersheds to larger streams with altered flow patterns due to dam construction. These aquatic habitats include tailwaters, impoundments, reservoir embayments, and large and small perennial streams.

A National Environmental Research Park Aquatic Reference Area is located along Grassy Creek and its tributaries, one of which runs through the eastern portion of the proposed spent nuclear fuel management site. Grassy Creek has a diverse assemblage of invertebrates and fish species for a stream its size. The Oak Ridge Reservation uses Grassy Creek as a reference area for studies of other streams affected by site development (Pounds et al., 1993).

Threatened, Endangered, and Candidate Plant and Animal Species: No animal species listed by the Federal Government as threatened or endangered are known to reside on the Oak Ridge Reservation (Kroodsma, 1987). The bald eagle is a winter visitor to Watts Bar Lake and Melton Hill Lake. None of the species listed in Table 4.9-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS have been recorded on the proposed West Bear Creek Valley site (DOE, 1995c). Table 4.9-1 of Appendix F, Volume 1 of the Programmatic SNF&INEL Final EIS lists Federally and State-listed threatened, endangered or other special-status species designated by the Endangered Species Act and/or the State's Nongame and Endangered Species and the Rare Plant Protection and Conservation laws (DOE, 1995c). Preferred habitat within the site indicates a greater potential for occurrence of the barn owl, black vulture, Cooper's hawk, red-shouldered hawk, and sharp-shinned hawk (DOE, 1995c). No intensive threatened and endangered species surveys have been completed for the site, but they are currently in progress for the entire the Oak Ridge Reservation (King et al., 1994).

3.3.4.6 Land Use

The Oak Ridge Reservation occupies an area of approximately 14,029 ha (34,667 acres) in eastern Tennessee, in a predominantly rural area about 40 km (25 mi) west of Knoxville. The Oak Ridge Reservation is within the jurisdictional boundaries of the city of Oak Ridge, and also lies within Roane and Anderson Counties (MMES, 1989).

Land use activities at the Oak Ridge Reservation have historically occurred within the boundaries of the three main plant sites (Y-12, the Oak Ridge National Laboratory, and K-25). The Oak Ridge Reservation has been used by research institutions, universities, and Government agencies as a site for the study of terrestrial ecology, aquatic ecology, forestry, and agriculture. Land uses bordering the Oak Ridge Reservation are primarily forest and agricultural. Residential and commercial are the only other significant uses of land in the vicinity, and occur along the northeast and northwest boundaries of the Oak

Ridge Reservation in the city of Oak Ridge. Figure 3-69 shows the land use in and around the Oak Ridge Reservation. The entire Oak Ridge Reservation has been placed under the forestry, agriculture, industry, and research zoning classification by the city of Oak Ridge. This classification does not bind DOE land use decisions on the site.

The region surrounding the Oak Ridge Reservation has numerous local, State, and national public recreation areas. Several lakes exist within the region surrounding the Oak Ridge Reservation, offering year-round recreational activities such as fishing and boating. The Oak Ridge Reservation is a controlled area, with public access limited to through traffic on Tennessee State Routes 95, 58, 62, 162, and 170 (MMES, 1991). There are no onsite areas that are subject to Native American Treaty rights or that contain any prime or unique farmland.

3.3.4.7 Noise

The major noise sources within the Oak Ridge Reservation occur primarily in developed operational areas and include various facilities, equipment, and machines. Major noise sources outside the operational areas consist primarily of vehicles and railroad operations. At the site boundary, noise from these sources would be barely distinguishable from background noise levels. The State of Tennessee has not established specific numerical environmental noise standards applicable to the Oak Ridge Reservation. The acoustic environment along the Oak Ridge Reservation site boundary is typical of a rural location, with the average soundlevel in the range of 35 to 50 decibels, A-weighted.

3.3.4.8 Transportation

The region of influence for the Oak Ridge Reservation includes site roads and regional roads up to approximately 24 km (15 mi) in Anderson, Blount, Knox, Loudon, and Roane counties. Primary roads on the Oak Ridge Reservation include Tennessee State Routes 95, 58, 62, 162, and 170 (Bethel Valley Road), and Bear Creek Road. Except for Bear Creek Road east of Route 95, all are public roads. The remaining roads on the Oak Ridge Reservation are private. Interstates 75 and 40, and Tennessee State Routes 162, 62, and 61 form a loop around the Oak Ridge Reservation (Figure 3-70).

Current baseline traffic (i.e., 1994) along segments providing access to the Oak Ridge Reservation is projected to contribute to differing service level conditions (TDOT, 1991). Tennessee State Route 61 would operate at level of service D between Interstate 75 at Norris and U.S. Route 25W at Clinton, and at level of service C between U.S. Route 25W at Clinton to Tennessee State Route 62 east of Oliver Springs. Tennessee State Routes 58 and 170 (providing access from the east), as well as Bear Creek Road, would operate between levels of service C and A. Tennessee State Routes 62 and 95 would operate at widely varying levels of service in the vicinity of the Oak Ridge Reservation. Tennessee State Route 62 would operate at a level of service E between Tennessee State Route 95 at Oak Ridge and Tennessee State Route 170 between Tennessee State Route 170 and Tennessee State Route 162. Tennessee State Route 95 would operate at a level of service E between Tennessee State Route 61 and Tennessee State Route 62 at Oak Ridge.

No public transportation service exists on the Oak Ridge Reservation. Other modes of transportation within the region of influence include railways and waterways. Railroad service in the region of influence is provided by CSX Transportation and the Norfolk Southern Corporation. Waterborne transportation is via the Clinch River. The Clinch River waterway has rarely been used for DOE business, and no designated port facilities exist for such purposes (U.S. Army Corps of Engineers, 1994).

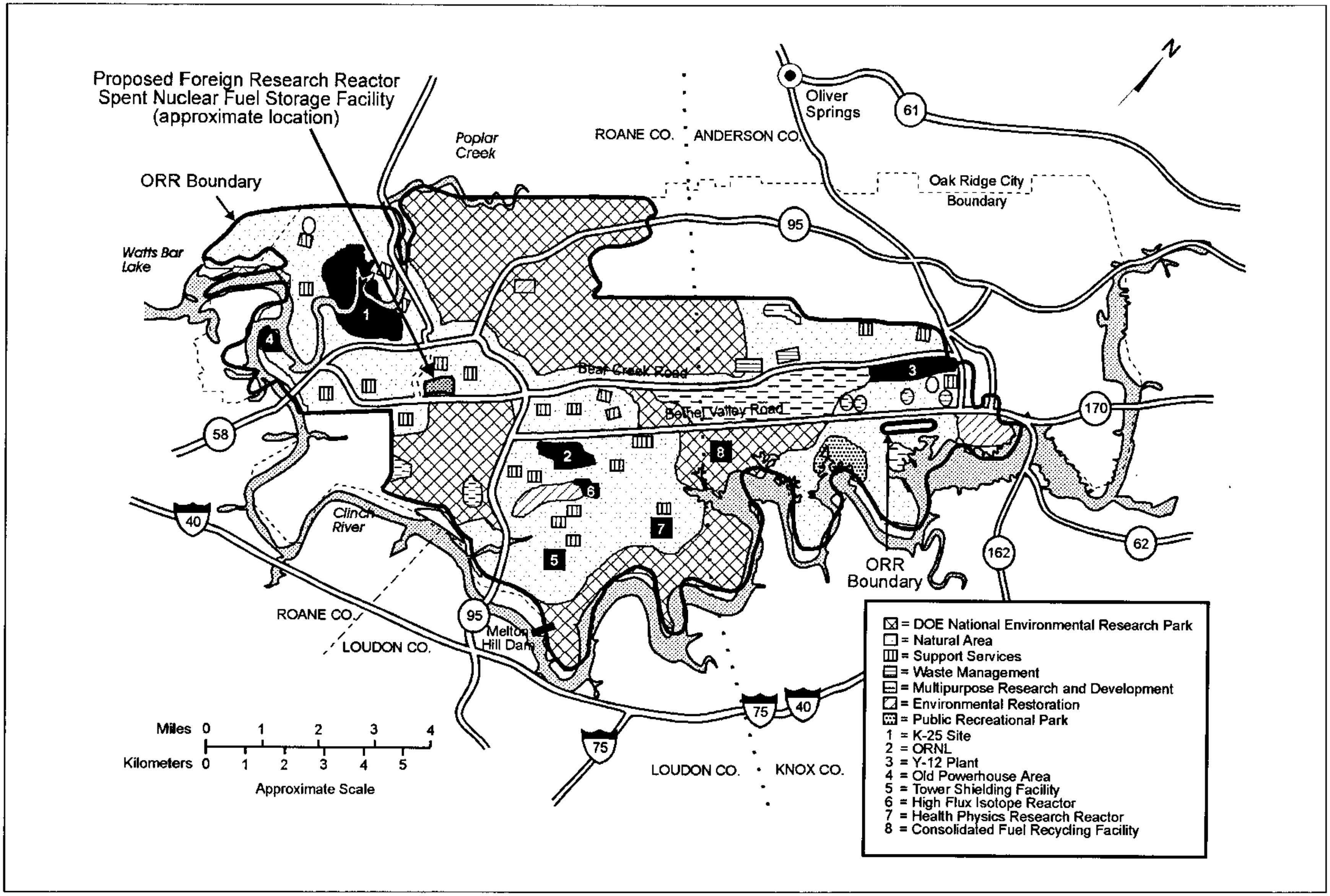


Figure 3-69 Generalized Land Use at the Oak Ridge Reservation

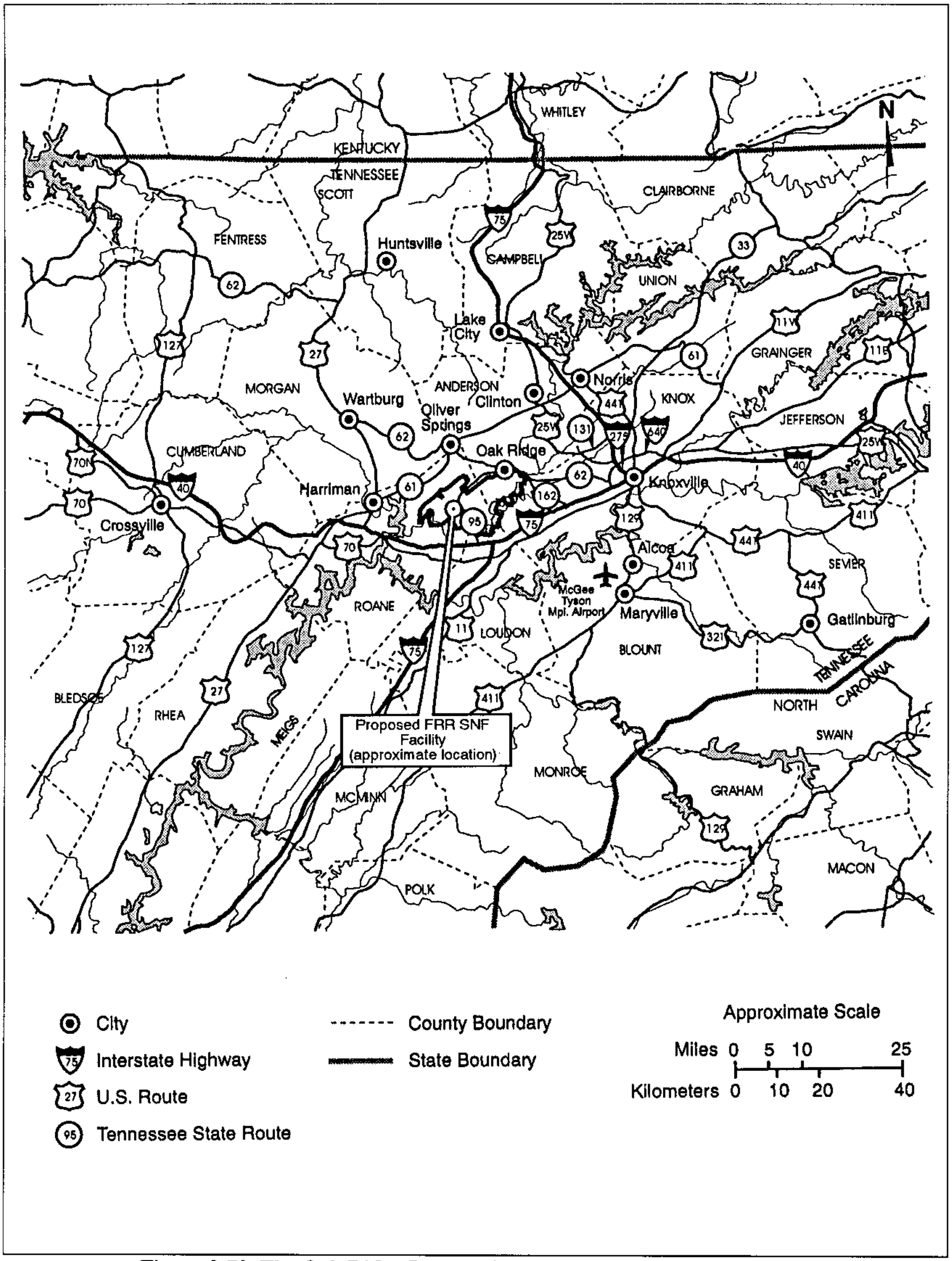


Figure 3-70 The Oak Ridge Reservation Regional Transportation Map

McGhee Tyson Airport in Knoxville, 64 km (40 mi) from the Oak Ridge Reservation, receives jet air passenger and cargo services from both national and international carriers. The closest air transportation facility to the Oak Ridge Reservation is Atomic Airport in Oliver Springs. Numerous other private airports are located throughout the region of influence (DOE, 1995c).

3.3.4.9 Socioeconomics

The region of influence includes the current residential distribution of DOE and contractor personnel employed by the Oak Ridge Reservation, the probable location of offsite contractor operations, and the probable location of labor and capital supporting indirect economic activity linked to the Oak Ridge Reservation. The region of influence includes the counties where 92 percent of DOE and contractor personnel employed by the Oak Ridge Reservation reside.

Regional economic linkage supporting production activity at the Oak Ridge Reservation occurs primarily with Anderson, Knox, Loudon, and Roane counties, where most of the offsite supporting contractors and labor and capital supporting indirect economic activity linked to the Oak Ridge Reservation are located. The total population of the region of influence is projected to be 489,230 persons in 1995 (DOE, 1995c), and is projected to grow at an annual average rate of less than 1 percent, reaching 538,820 persons in 2004. The labor force is also projected to grow at an annual average rate of less than 1 percent, growing to 360,000 persons in 2004. The total employment is projected to grow at an annual average rate of approximately 1 percent, growing from 292,700 jobs in 1995 to 338,070 jobs in 2004.

Figure 3-71 shows the racial and ethnic composition of minorities within 80 km (50 mi) of spent nuclear fuel management sites on the Oak Ridge Reservation. In comparison with the other four candidate sites, the Oak Ridge Reservation has the smallest percentage, about 6 percent, of minorities in the population residing around the site. African Americans make up approximately 76 percent of the minority population, while Hispanics and Asian Americans make up 8 to 9 percent of the minority population.

Figure 3-72 presents the low-income households residing within 80 km (50 mi) of the Oak Ridge Reservation. About 44 percent of the households are classified as having an income no larger than 80 percent of the median income for the county of residence. This percentage is typical of that for counties within 80 km (50 mi) of the spent nuclear fuel management sites.

The Oak Ridge Reservation fire protection services are provided by the fire departments on the reservation. The Oak Ridge Reservation fire departments have mutual aid agreements among themselves and with the city of Oak Ridge (DOE, 1995c). Twelve city, county, and State law enforcement agencies provide police protection in the region of influence. Law enforcement on the Oak Ridge Reservation is provided by the city of Oak Ridge Police Department. Security enforcement is provided by the prime management and operations contractor (MMES, 1989). Four county school districts (Anderson, Knox, Loudon, and Roane) and five city school districts (Clinton, Oak Ridge, Lenoir City, Kingston, and Harriman) provide public education services in the region of influence. In 1992, the nine school districts had an average daily membership of 75,825 students. Between 1980 and 1990, the number of housing units in the region of influence increased 14 percent from 181,299 to 206,234. In 1980 and 1990, the homeowner vacancy rates in the region of influence averaged 1.4 and 1.5 percent, respectively (DOE, 1995c).

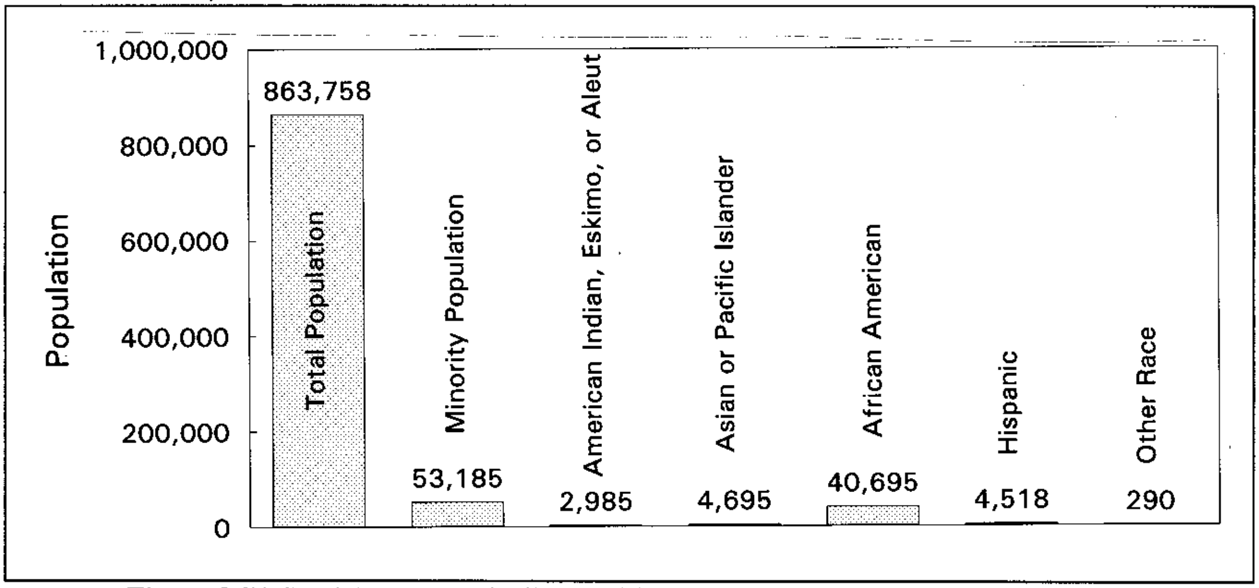


Figure 3-71 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Oak Ridge Reservation

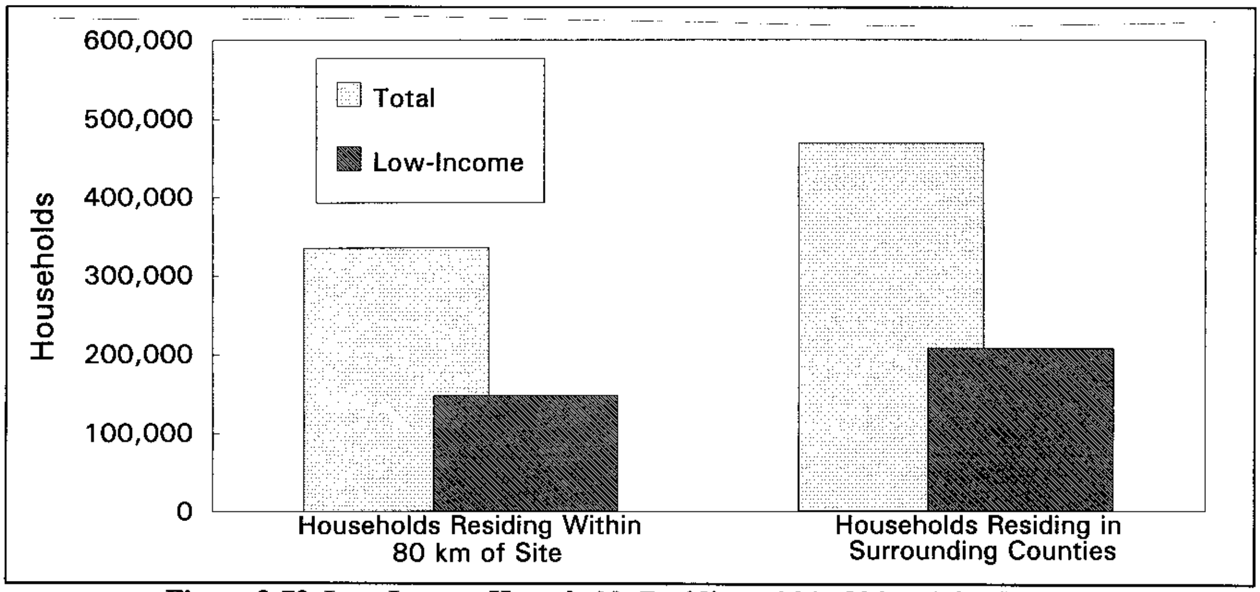


Figure 3-72 Low-Income Households Residing within 80 km (50 mi) of the Oak Ridge Reservation

3.3.4.10 Historical, Archaeological, and Cultural Resources

A limited survey conducted in 1975 did not identify any cultural resources on the Oak Ridge Reservation at the West Bear Creek Valley site (DOE, 1995c). No prehistoric or historic resources are expected to be located on the site. There are no known Native American resources on the proposed site.

3.3.5 Description of the Affected Environment at the Nevada Test Site

The Nevada Test Site is primarily used for the development and testing of nuclear weapons. This section describes the potentially affected environment of the site. The location of the site is shown in Figure 3-73.

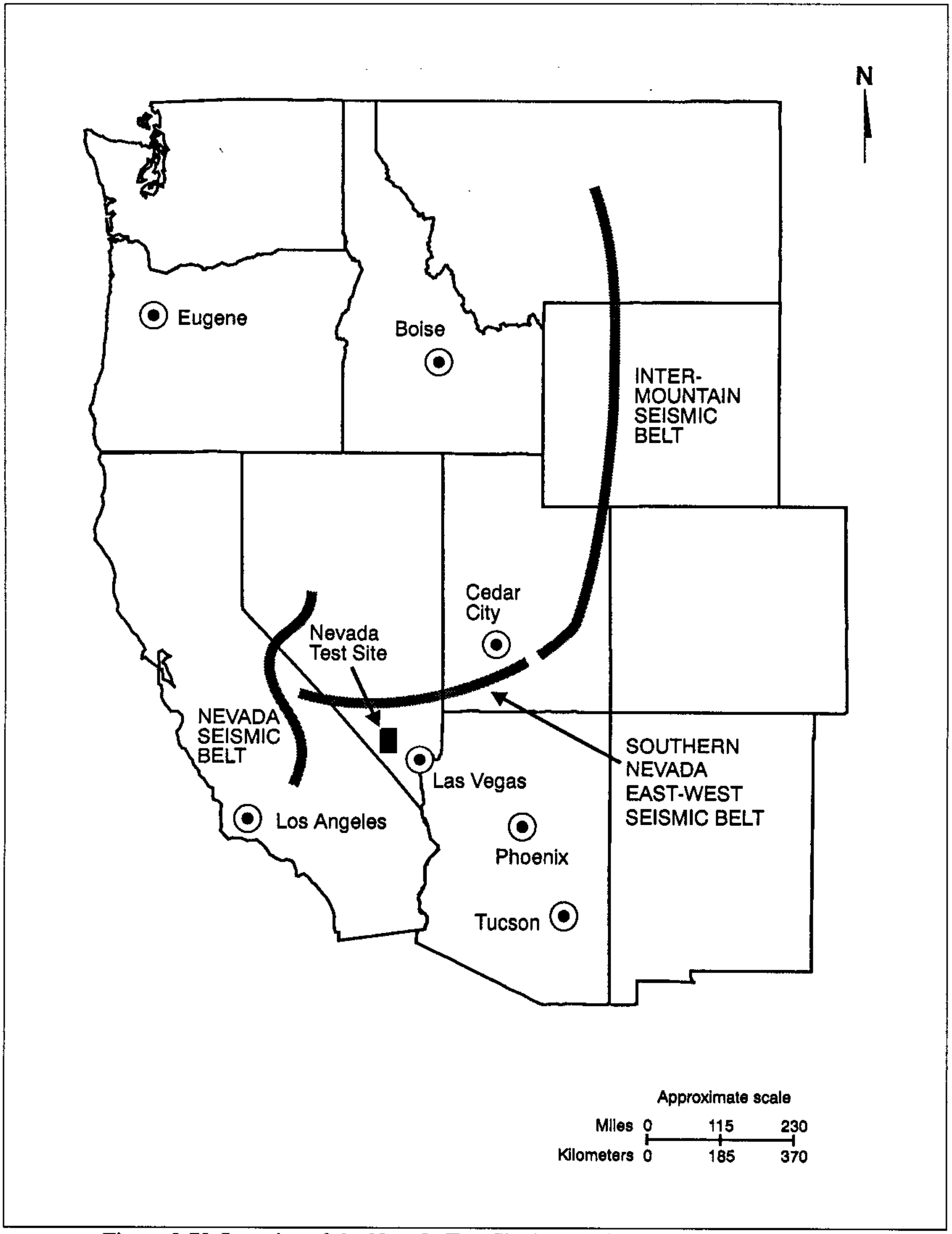


Figure 3-73 Location of the Nevada Test Site in Relation to the Nevada Seismic Belt, the Intermountain Seismic Belt, and the Southern Nevada East-West Seismic Belt

3.3.5.1 Geology

The Nevada Test Site is located east and north of the Walker Lane-Las Vegas Valley Shear Zone (Eckel, 1968). Walker Lane is a northwest-trending belt of right-lateral faults that disrupts the regional structural grain in the southwestern part of the Great Basin along the California-Nevada border. The Las Vegas Valley Shear Zone is a concealed zone of right-lateral faulting along the north side of the Las Vegas Valley (DOE, 1988a). The local geology of the Nevada Test Site is characterized by mountain ranges composed of Precambrian and Paleozoic sedimentary rocks and Tertiary volcanic tuffs and lavas that surround alluvium-filled, topographically closed valleys. A geologic map of the site is shown as Figure 3-74 (DOE, 1993b). The sedimentary rocks are complexly folded and faulted, and are composed mainly of carbonates (dolomite and limestone) in the upper and lower parts of the column and clastics (shale and sandstone) in the middle section. The volcanic rocks are predominantly tuffs that are high in silica.

Faulting generally occurs as thrust faults, normal faults, and strike-slip faults (DOE, 1992c). The faults are shown in Figure 3-75 (DOE, 1993b). Thrust faulting occurs as three major thrust faults, and normal faults exist in both ranges and valleys and generally strike northeast and northwest. The nearest strike-slip structure to the Nevada Test Site is the Walker Lane-Las Vegas Valley Shear Zone. Estimates of horizontal displacement along this shear zone range from 40 to 160 km (25 to 100 mi) (DOE, 1982). Recent displacements have occurred along several faults as a consequence of underground nuclear explosions. This disturbance is not attributable to naturally-occurring seismic activity. Fault displacements are thought to have occurred as a result of the added stress produced by the explosions, the vibrations produced by the explosions, or a combination of both (Eckel, 1968). Almost all of the natural fault movement in the Nevada Test Site area occurred several million years ago, except movement along Yucca Fault. Movement in the Yucca Fault is believed to have occurred in the past tens of thousands to 250,000 years (DOE, 1982; DOE, 1995c).

3.3.5.2 Seismology

The Nevada Test Site lies on the southern margin of the Southern Nevada East-West Seismic Belt (Figure 3-73), which is an area of relatively low historical seismicity. The regional seismicity is dominated by high-levels of seismic activity. Between 1978 and 1981, no earthquakes with magnitudes greater than 4.3 were recorded (DOE, 1986b). In 1992, a magnitude 5.6 earthquake was recorded near Little Skull Mountain at a depth of 12 km (7.5 mi). In 1993, a magnitude 3.5 earthquake was recorded southeast of the town of Mercury on the Nevada Test Site (DOE, 1995c).

The most probable source for seismic activity at the Nevada Test Site is the Cane Spring fault (Figure 3-75). Estimates of recurrence intervals for major earthquakes in the region are on the order of 25,000 years. For magnitudes of greater than or equal to 6, recurrence intervals are on the order of 2,500 years, and for magnitudes of greater than or equal to 5, recurrence intervals are on the order of 250 years (DOE, 1986b).

3.3.5.3 Hydrology

3.3.5.3.1 Surface Water

The drainage basins and the generalized directions of surface water flow near the Nevada Test Site are shown in Figure 3-76 (USAF et al., 1991). The boundary lines of the drainage basins occur principally along topographic divides (DOE, 1988a). Almost all streamflow in the Nevada Test Site area is

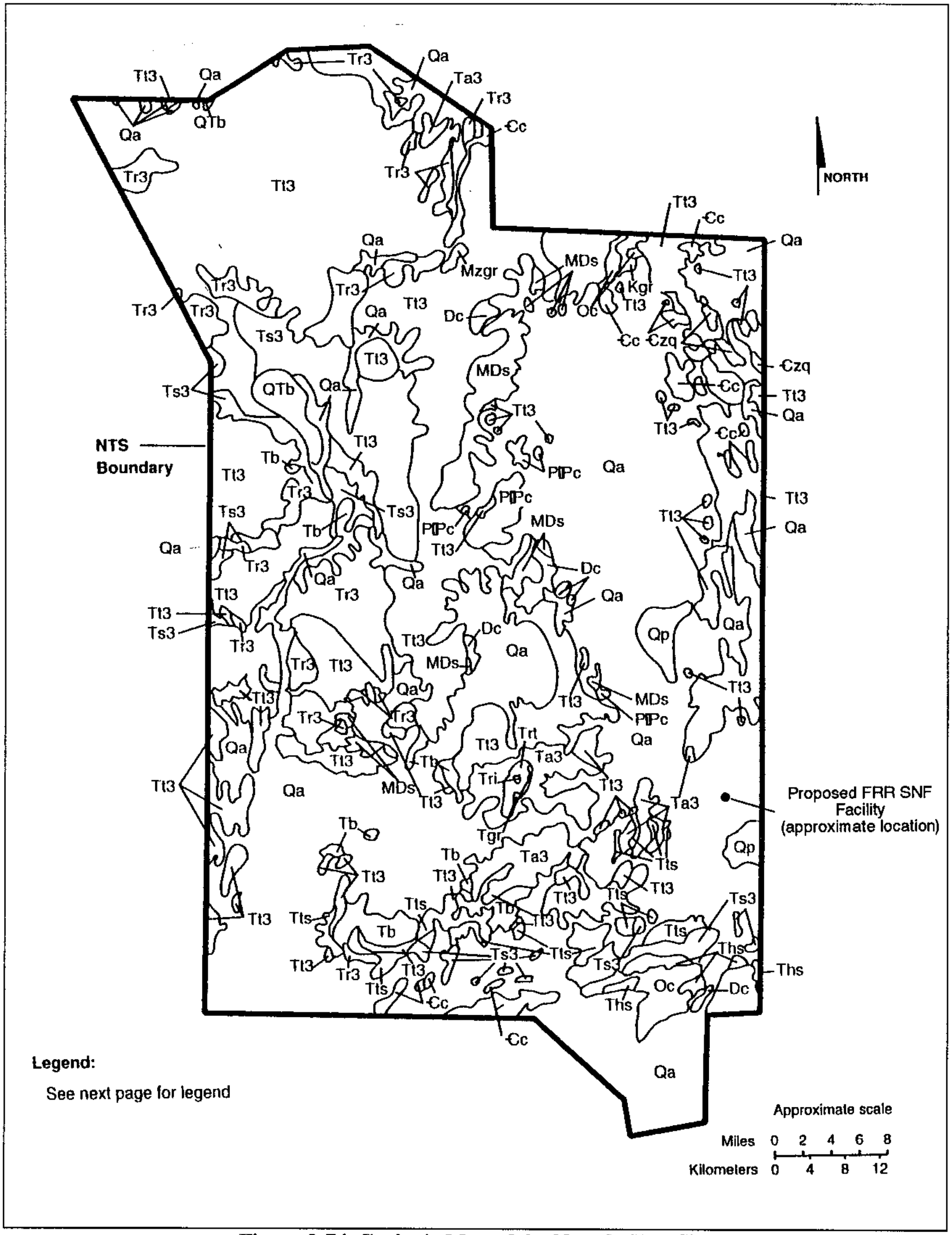


Figure 3-74 Geologic Map of the Nevada Test Site

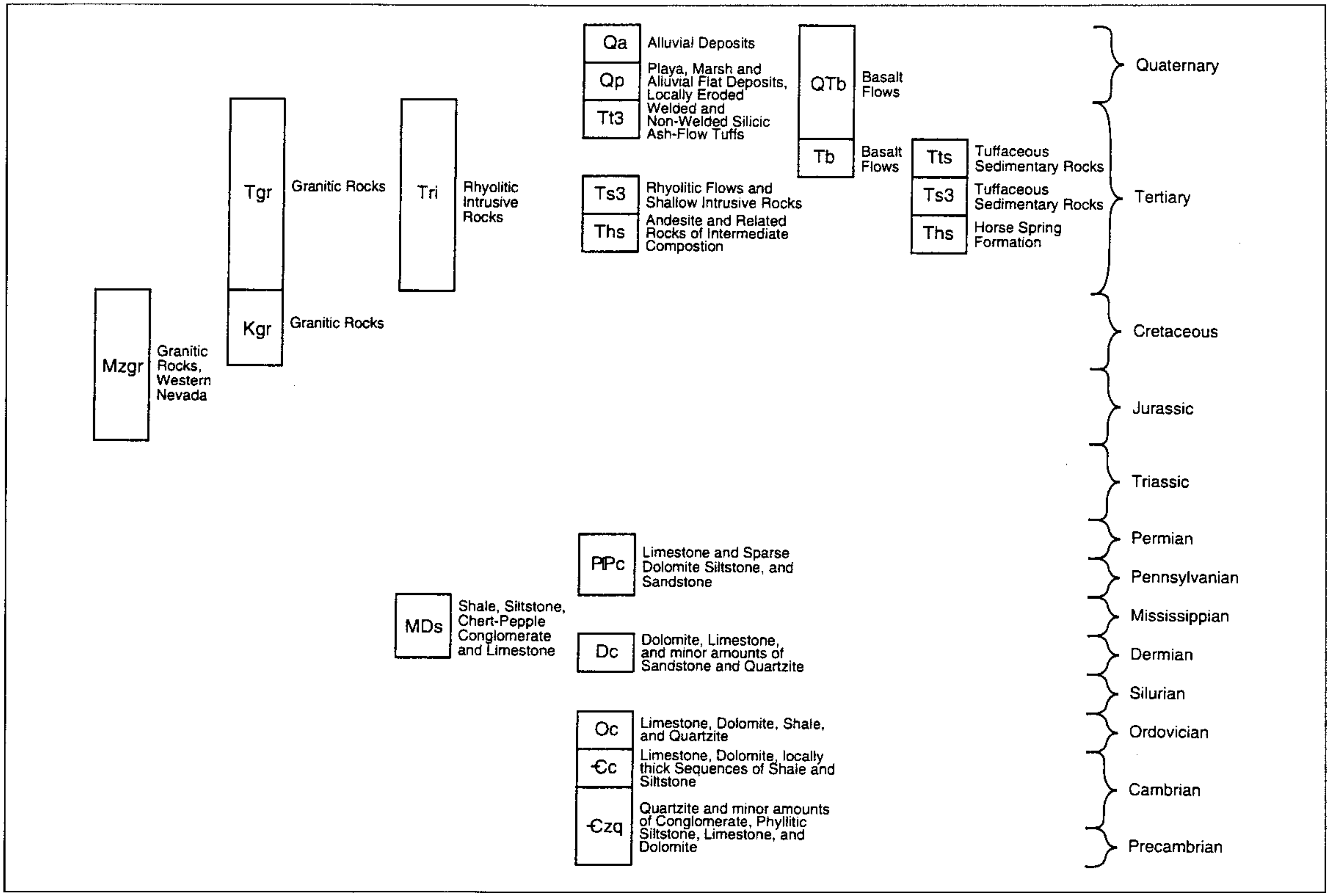


Figure 3-74 Geologic Map of the Nevada Test Site (Continued)

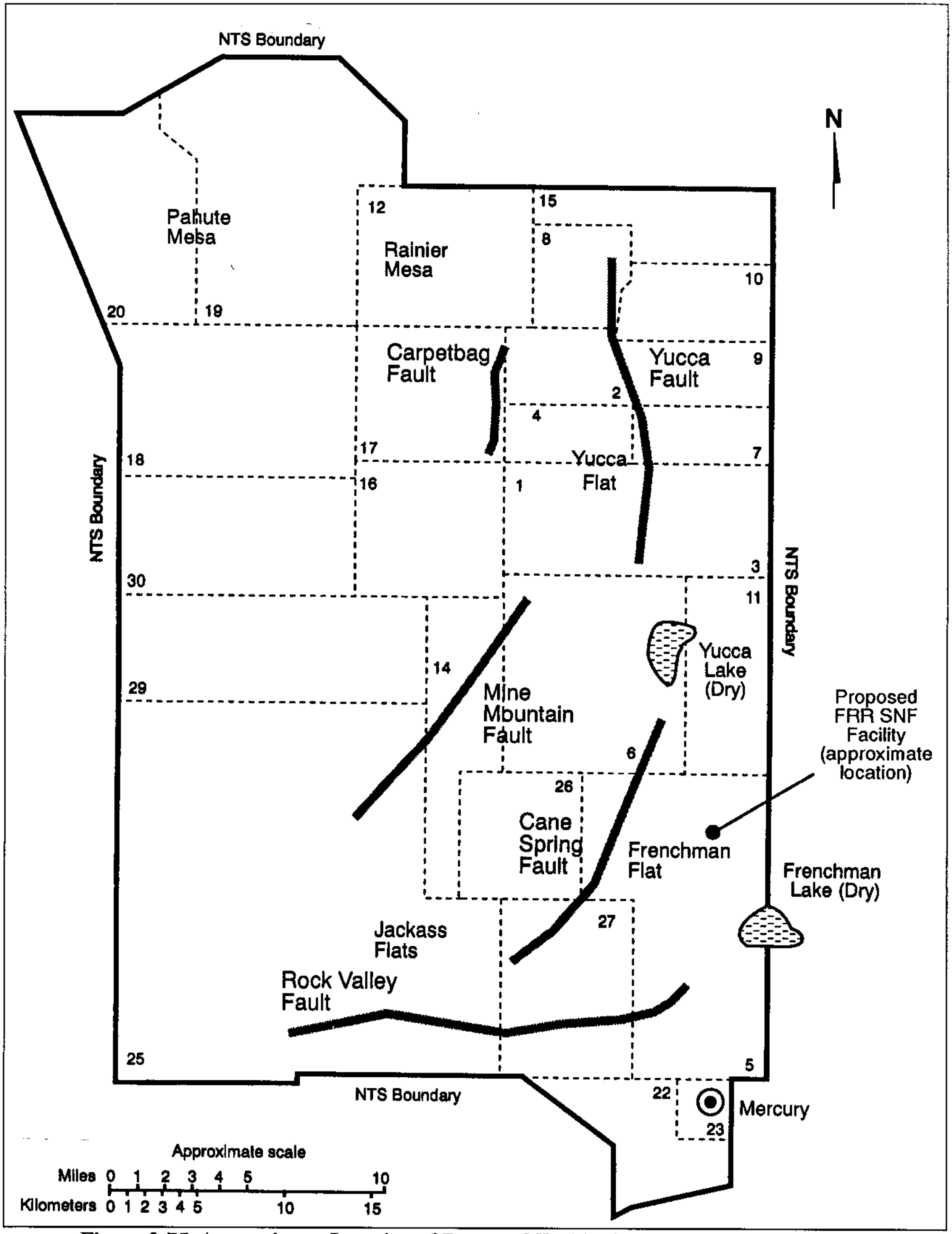


Figure 3-75 Approximate Location of Proposed Facility in Relation to Major Faults at the Nevada Test Site

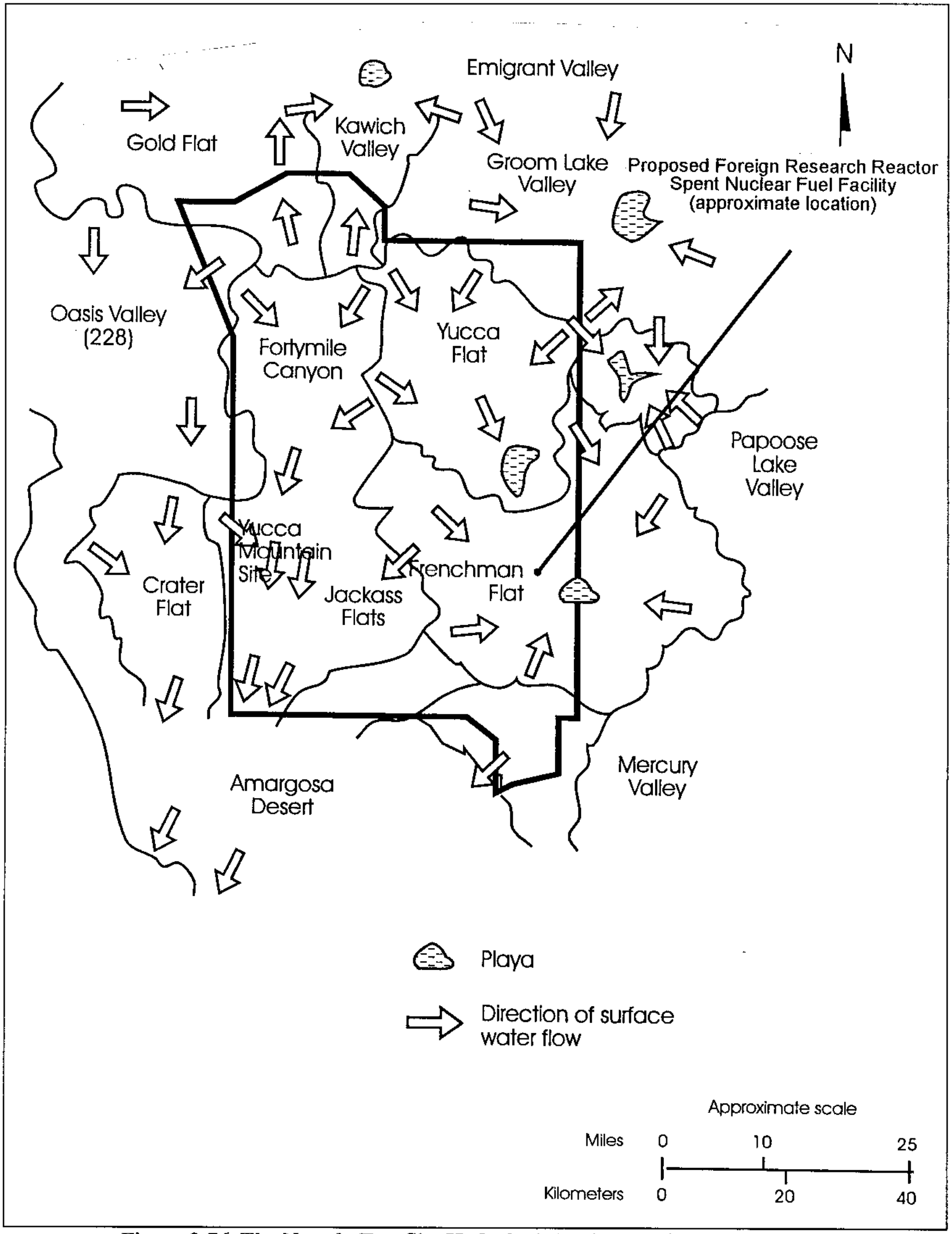


Figure 3-76 The Nevada Test Site Hydrologic Basins and Surface Drainage Direction

ephemeral, and therefore almost no streamflow data have been collected. The ephemeral character of streamflow has also limited the onsite monitoring of surface water quality. Perennial surface water originates from springs, and is restricted to source pools at some large springs. Because of the extreme aridity of this region, most of the spring discharge travels only a short distance before evaporating or infiltrating back into the ground (DOE, 1986b). The western half and southernmost part of the Nevada Test Site have channel systems which carry runoff beyond the Nevada Test Site boundaries during infrequent, very intense storms.

Two watersheds, Fortymile Canyon and Jackass Flats, have the potential for endangering offsite public health and safety due to flooding. Regional peak-flood flow equations for the southern Nevada area indicate that the 100-yr peak flow from the Fortymile Canyon drainage is approximately 370 m^3 per sec (97,744 gal per sec), and 230 m^3 per sec (60,760 gal per sec) from the Jackass Flats drainage (USAF et al., 1991). Underground nuclear testing has resulted in the release of radioactive materials at the land surface. There is the potential for 100-yr floods to transport these contaminants beyond the boundaries of the Nevada Test Site.

3.3.5.3.2 Groundwater

The hydrogeology at the Nevada Test Site is characterized by great depths to the groundwater table, and slow velocity of movement of water in the saturated and unsaturated zones (DOE, 1992c). Depth to groundwater varies from about 200 m (660 ft) beneath valleys in the southern part of the Nevada Test Site, to more than 500 m (1,650 ft) beneath Pahute Mesa. Locally, there are perched water tables at shallow depths (USAF et al., 1991). Perched aquifers have been reported at depths of 21 m (70 ft) in the southwestern part of Frenchman Flat. In the eastern portions of the Nevada Test Site, the water table occurs generally in the alluvium and volcanic rocks above the regional carbonate aquifer (DOE, 1993a).

Six major aquifers occur in the area. The hydrologic and geologic properties of these aquifers vary (DOE, 1988a). The lower carbonate and valley fill aquifers are the main sources of groundwater in the eastern part of the Nevada Test Site (DOE, 1986b). Four major aquitards (units tending to retard the flow of groundwater) have been reported in the area (DOE, 1986b).

Regional groundwater flow is from the north and northeast toward the regional discharge area near Ash Meadows in the Amargosa Desert (Figure 3-77). In the western portions of the area, the regional flow is from the northwest to the south and southwest. Groundwater recharge to the Ash Meadows Sub basin occurs primarily from precipitation over the mountainous areas in the northern, eastern, and southern portions of the basin (DOE, 1988a).

The hydrogeologic units that supply potable water to the Nevada Test Site have been classified as Class IIA (currently a source of drinking water) and IIB (potentially a source of drinking water), in accordance with the U.S. Environmental Protection Agency's guidelines for groundwater classification. No aquifers at the Nevada Test Site have been designated as sole-source aquifers.

Groundwater Quality: The quality of the Nevada Test Site groundwater is suitable for most purposes, and generally meets U.S. Environmental Protection Agency secondary standards for major cations and anions, and the primary standards for deleterious constituents. Contamination by radionuclides occurs below the water table as well as in the unsaturated zone above it. This contamination is a result of underground nuclear testing. The principal confirmed or suspected contaminants from these tests include various radionuclides (primarily tritium) and heavy metals.

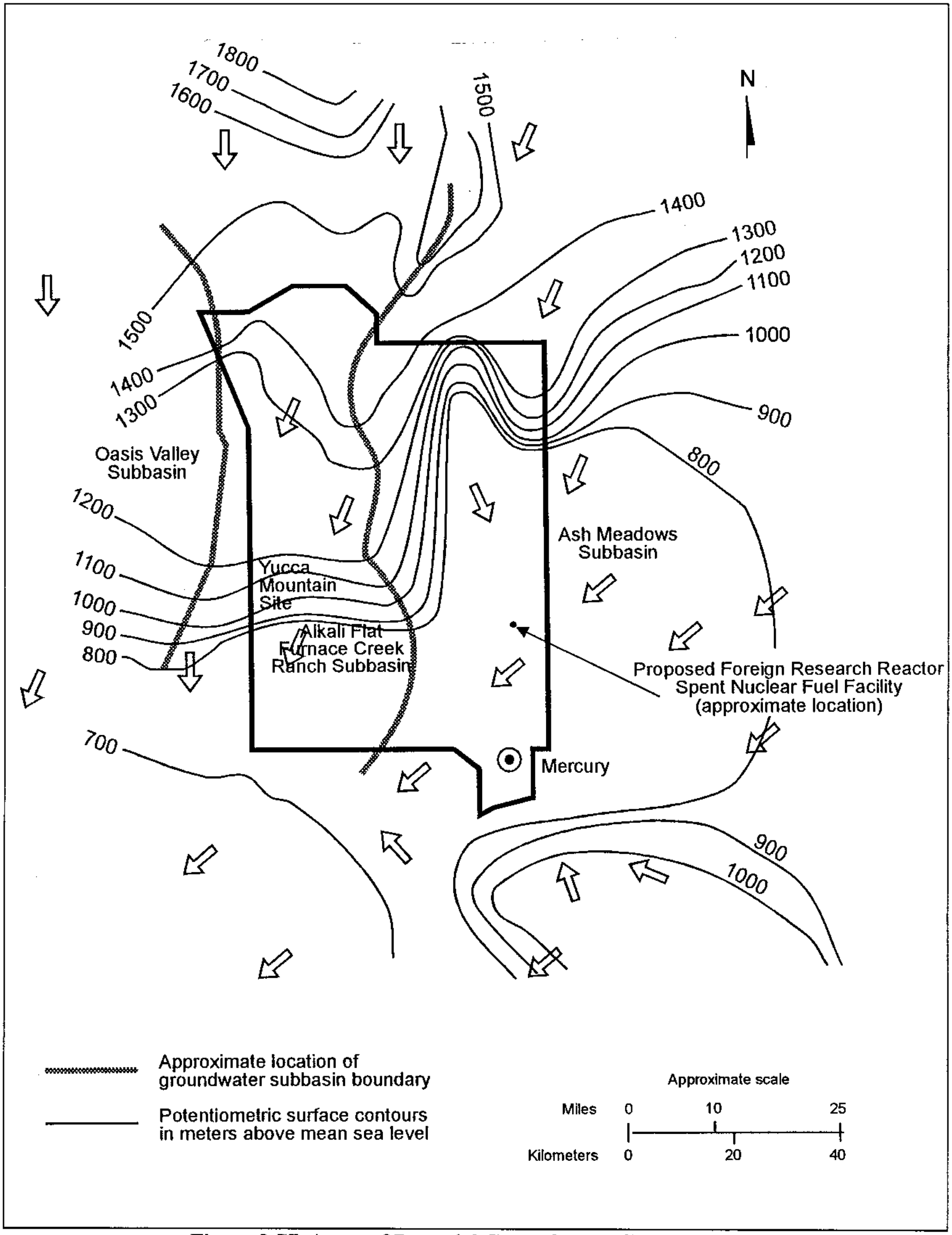


Figure 3-77 Areas of Potential Groundwater Contamination

Groundwater contamination could be transported toward the Nevada Test Site boundary by one of the regional groundwater flow systems. Groundwater flow velocities in these systems range between 1.8 and 183 m per year (6 and 600 ft per year). Due to sorption, most nuclides (other than tritium) would move at a much slower rate. The groundwater travel time from the Nevada Test Site to Ash Meadows Discharge Area is approximately 300 years. Radioactive decay, coupled with dilution and sorption, should reduce radioactivity to well below regulatory limits (USAF et al., 1991). Thus, there are no current effects on public health and safety, nor are any expected in the foreseeable future.

3.3.5.4 Meteorology

The climate at the Nevada Test Site and the surrounding region is characterized by high solar radiation, limited precipitation, low relative humidity, and large diurnal temperature ranges. The lower elevations have a climate typical of the Great Basin.

Wind: Low-level surface winds at the Nevada Test Site are influenced by the large-scale weather patterns interacting with the mountain ranges. Predominant winds are from the south during the summer, and the north during the winter. At Las Vegas, the peak wind gust on record is 145 km per hour (90 mph). Strong winds interacting with dry soil conditions are responsible for occasional duststorms or sandstorms.

Temperature and Humidity: At Area 6 (Figure 3-75) of the Nevada Test Site, the average daily maximum/minimum temperatures during the month of January are 10.6°C/-6.1°C (51.1°F/21.0°F). The average daily maximum/minimum temperatures are 35.6°C/13.9°C (96.1°F/57.0°F) in July. At Las Vegas, the coldest temperature on record is -13.3°C (8.1°F), and the warmest temperature on record is 46.7°C (116°F). The average relative humidity at 4 a.m. in Las Vegas is 40 percent. The average relative humidity at 4 p.m. is 20 percent (DOE, 1995c).

Precipitation: The average annual precipitation at Area 6 is 15 cm (5.9 in). Precipitation amounts for each month are generally less than 1.3 cm (.5 in). At Las Vegas, the greatest precipitation recorded in a 24-hr period is 6.6 cm (2.6 in). An average of 14 thunderstorm days occur each year, with maximum occurrence in July and August. Thunderstorms occasionally become severe. Tornadoes are extremely rare in Nevada (DOE, 1995c).

Atmospheric Dispersion: Data collected at Desert Rock for calendar year 1990 indicated that atmospheric conditions were unstable (Stability Classes A through C) approximately 25 percent of the time, neutral (Class D) approximately 37 percent of the time, and stable (Classes E through G) approximately 37 percent of the time for that year (DOE, 1995c).

Air Quality: In 1992, the majority of radioactive effluents at the Nevada Test Site originated from underground nuclear tests designed and conducted by two national laboratories and the Defense Nuclear Agency. Onsite monitoring of airborne particulates, noble gases, and tritiated water vapor indicated onsite concentrations that were generally not statistically different from background concentrations. Results of offsite environmental monitoring indicated none of the Nevada Test Site-related radioactivity was detected at any air sampling station, and there were no apparent net exposures detectable by the offsite dosimetry network (DOE, 1993a).

The nonradiological air emissions from the Nevada Test Site originate from concrete batch plants, aggregate crushing and processing, surface disturbance, fire training exercises, motor vehicle operations, boilers, and fuel storage. Based on the data collected by Engineering Science, Inc. at the ambient air monitoring stations, air quality at the Nevada Test Site is within applicable Federal and State standards (DOE, 1995c).

3.3.5.5 Ecology

The Nevada Test Site lies within the transition area of the Mojave desert and the Great Basin. The Nevada Test Site covers about 3,500 km^2 (1,350 mi^2), of which only 0.55 percent is developed (DOE, 1988b).

Plant communities on the Nevada Test Site have been classified according to the dominant shrub. Approximately 700 taxa have been identified on the Nevada Test Site (ERDA 1976; DOE, 1991b, DOE, 1993b). Figure 3-78 presents the general plant communities identified on the Nevada Test Site. The dominant plant communities in the Mojave desert are creosote bush. The dominant plant communities in the transition zone between the Mojave desert and the Great Basin are blackbrush, desert thorn, and hopsage. The dominant plant communities in the Great Basin are big sagebrush and black sagebrush, saltbush, and desert thorn.

There are more than 30 species of reptiles and amphibians, 190 species of birds, and 50 species of mammals on the Nevada Test Site (ERDA, 1976; RSN, 1993). Sewage ponds and man-made reservoirs have become an important resource for wildlife. Reptiles and amphibians on the Nevada Test Site include 1 species of desert tortoise, 14 species of lizards, and 17 species of snakes. Birds on the Nevada Test Site are often migratory and seasonal residents. The most-distributed species include the black-throated sparrow, house fin, red-tailed hawk, common raven, loggerhead shrike, mockingbird, ash-throated flycatcher, and mourning dove (Greger, n.d.a.; ERDA, 1976). The most abundant group of mammals on the Nevada Test Site are rodents.

There are several natural springs on the Nevada Test Site that feed flowing streams (Greger, n.d.a.). Vegetation along these channels consists of willow and tamarisk. National Wetlands Inventory maps are not available for the Nevada Test Site (DOE, 1995c).

Potential aquatic habitat on the Nevada Test Site includes surface drainage, playas, man-made reservoirs, and springs. Permanent surface water resources are limited to a few small springs. These surface drainage and playas are unable to support permanent fish populations (ERDA, 1976).

Threatened, Endangered, and Candidate Plant and Animal Species: Table 4.9-1 of Appendix F, Volume 1 of the SNF&INEL Final EIS presents a list of Federally and State-listed species that may be found in the vicinity of the Nevada Test Site (DOE, 1995c). There are no known plants that have been listed as threatened or endangered under the Endangered Species Act on the Nevada Test Site. However, the U.S. Fish and Wildlife Service has identified candidate species for listing, 11 of which may occur on or in the vicinity of the Nevada Test Site. Ten of these are Category 2 species (may be appropriate for listing as endangered or threatened but more information is needed).

Two listed reptile species on or in the vicinity of the Nevada Test Site are of concern. The chuckwalla is a Federal Candidate Category 2 species which may occur on the Nevada Test Site. The desert tortoise is the only Federally listed threatened species known to occur on the Nevada Test Site (DOE, 1995c).

Two bird species (American peregrine falcon and the bald eagle) which could occur on or within the vicinity of the Nevada Test Site are Federally listed endangered species. There are two (spotted bar and pygmy rabbit) Federal Candidate Category 2 mammal species identified as potentially occurring in the vicinity of the Nevada Test Site. There are no known fish species indigenous to the Nevada Test Site.

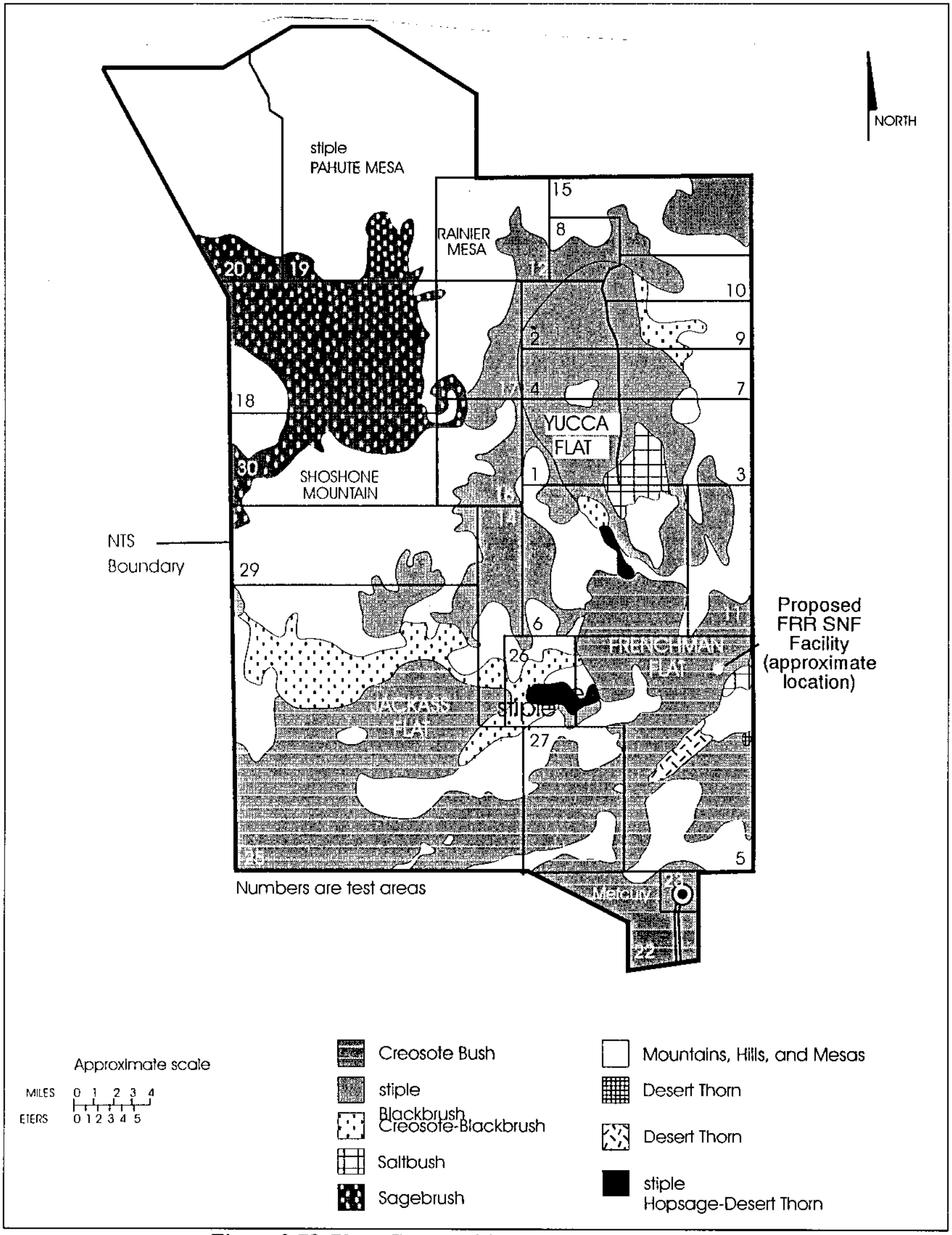


Figure 3-78 Plant Communities on the Nevada Test Site

3.3.5.6 Land Use

The Nevada Test Site occupies an area of approximately 3,500 km^2 (1,350 mi^2) in southern Nevada, in a sparsely populated area approximately 105 km (65 mi) northwest of Las Vegas. The Nevada Test Site is almost entirely surrounded by other Federally-owned lands that buffer it from lands open to the public. The Nevada Test Site is bordered by the Nellis Air Force Range on the north, east, and west, and by Bureau of Land Management lands on the south and southwest (DOE, 1993b).

Existing land use on the Nevada Test Site falls into four general categories: Testing Areas, Buffer/Reserved Areas, Industrial/Research Areas, and Waste Management Areas. According to the latest the Nevada Test Site land use map (Figure 3-79), approximately 50 percent of the land on the Nevada Test Site is buffer/reserved area for ongoing programs or projects (DOE, 1993a).

The Nevada Test Site is located in an area of sparsely vegetated desert. Principal uses in Nye County in the vicinity of the Nevada Test Site include mining, grazing, agriculture, and recreation (DOE, 1993a). Urban and residential land uses occur beyond the immediate vicinity of the Nevada Test Site. Clark County, to the southeast of the Nevada Test Site, consists of approximately 20,460 km^2 (7,900 mi^2), of which about 95 percent is owned by the Federal Government.

Numerous national, State, and local public recreation areas exist within the Nevada Test Site region. The Nevada Test Site is a controlled area, with public access limited to through traffic on U.S. Route 95, and on Lathrop Wells Road (DOE, 1993a). There are no onsite areas subject to Native American Treaty rights or that contain any prime or unique farmland (PIC, 1992).

3.3.5.7 Noise

The major noise sources at the Nevada Test Site occur primarily in developed operational areas, and include various facilities, equipment and machines, aircraft operations, and testing. No Nevada Test Site environmental noise survey data are available (DOE, 1995c). At the Nevada Test Site boundaries, noise from most sources is barely distinguishable from background noise levels. Transportation of people and materials to and from the Nevada Test Site is the noise source of importance to the public. During a normal work week about 3,300 employees travel to the Nevada Test Site each day (DOE, 1995c).

3.3.5.8 Transportation

Vehicular access to the Nevada Test Site is provided by U.S. Route 95 to the south, with off-road access to the northeast provided via Nevada State Route 375. Nevada State Route 375 and U.S. Route 95 are projected to remain at Level of Service A (free flow of traffic). The public transit serves the populated regions of Clark County. Contract buses run to the Nevada Test Site. There is no public transportation system serving the Nevada Test Site, but 70 buses a day transport employees to and from the site.

The nearest railroad is the Union Pacific, located approximately 80 km (50 mi) east of the Nevada Test Site near Las Vegas. No navigable waterways are capable of accommodating waterborne transportation of material shipments to the Nevada Test Site. McCarran International Airport in Las Vegas provides jet air passenger and cargo service from both national and local carriers.

3.3.5.9 Socioeconomics

A Nevada Test Site worker residential distribution survey from 1988 indicates that 88 percent lived in Clark County and 10 percent in Nye County (DOE, 1995c). In Clark County, most of the Nevada Test Site employees reside in the Las Vegas vicinity.

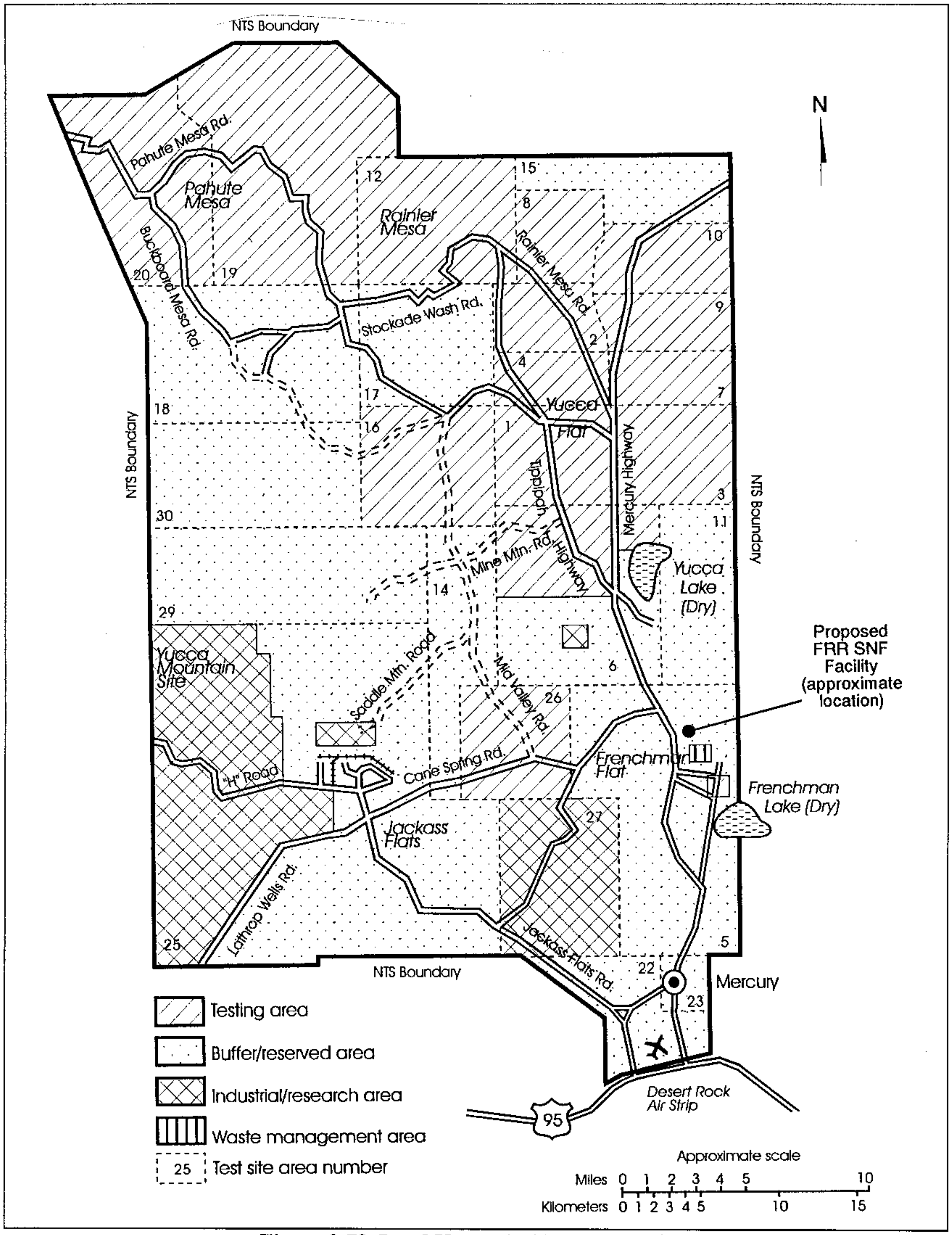


Figure 3-79 Land Use at the Nevada Test Site

Clark County: Clark County is composed of five incorporated cities (Las Vegas, Henderson, North Las Vegas, Boulder City, and Mesquite), and large expanses of unincorporated land. The area experiencing the majority of the county's development is the Las Vegas Valley (DOE, 1995c). Economic conditions in southern Nevada have improved continuously since the mid-1980's. The economy is driven by the growth in the hotel and gaming industry. Service employment accounts for nearly 45 percent of total employment, trade employment accounts for 21 percent, and Government and construction each account for an additional 10 percent (DOE, 1995c).

The unemployment rate reached a low of 4.9 percent in 1990, and increased to 7.5 percent as of June, 1993. However, the unemployment level is expected to decrease with new hotel, gaming, and amusement properties which opened at the end of 1993 (DOE, 1993a).

Nye County: The employment level in Nye county is low relative to Clark County, and includes opportunities in the services, mining, and Government sectors (DOE, 1993a). Nye County is sparsely populated, with the two largest population groupings in the communities of Pahrump and Tonopah. While tourist activity is an important part of the Nye County economy in communities along U.S. 95, mining is the major, even dominant, economic force.

The Nevada Test Site: The Nevada Test Site work force supports engineering design, construction, and operation of the site. As of January 1994, there were a total of 8,563 (3,286 on Nevada Test Site, 3,805 in Las Vegas, and 1,472 in the rest of Nevada or other areas) . The population within the 80 km (50 mi) radius of the Nevada Test Site is approximately 12,421. Minority population constitutes approximately 16 percent of the total. Figure 3-80 shows the racial and ethnic composition of the minority population within 80 km (50 mi) of the Nevada Test Site. Hispanics form more than 50 percent of the minority population.

The general characteristics of the low-income households residing within 80 km (50 mi) of the Nevada Test Site are presented in Figure 3-81. Low-income households are 48 percent of the total households.

The Nevada Test Site's fire protection capacity is structured to accommodate current mission requirements, with a self-contained firefighting department responsible for suppression and prevention. Other services include rescue, hazardous material response, training of fire personnel, fire prevention inspections, installation of all fire extinguishers at the Nevada Test Site, and fire prevention awareness programs. There is a mutual agreement between the Clark County Fire Department and all surrounding area departments to assist in any fire emergency when necessary (DOE, 1993a).

Health Care: The Nevada Test Site has a self-contained medical center that provides limited emergency treatment. Health care in the Las Vegas Area is provided through 13 full-service hospitals, with 3.44 hospital beds per 1,000 members of the population.

Education and Training: The Clark County school district provides education services for the employees who work at the Nevada Test Site. An average student/teacher ratio of 22:32 is reported for elementary school grades K-6 (DOE, 1993a). There are a number of vocational, training, and higher education institutions in the Las Vegas metropolitan area (DOE, 1993a).

Housing: Between 1980 and 1990, the number of housing units in Clark County increased by 84 percent. The increase in demand is attributable to the influx of retirees and other in-migrant population.

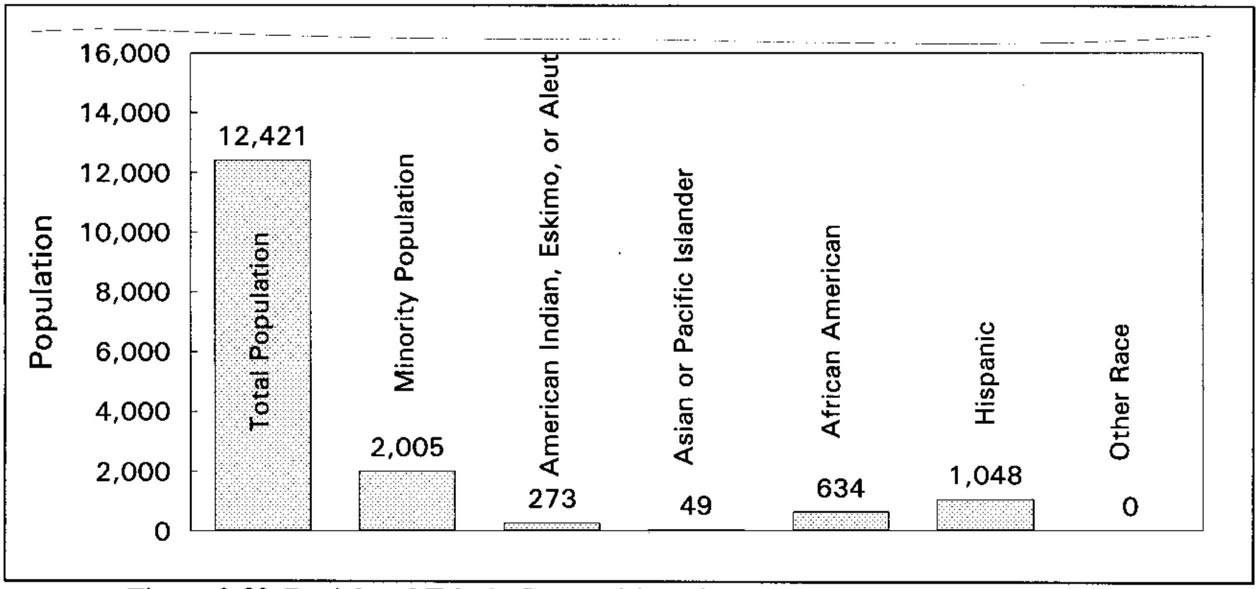


Figure 3-80 Racial and Ethnic Composition of the Minority Population Residing within 80 km (50 mi) of the Nevada Test Site

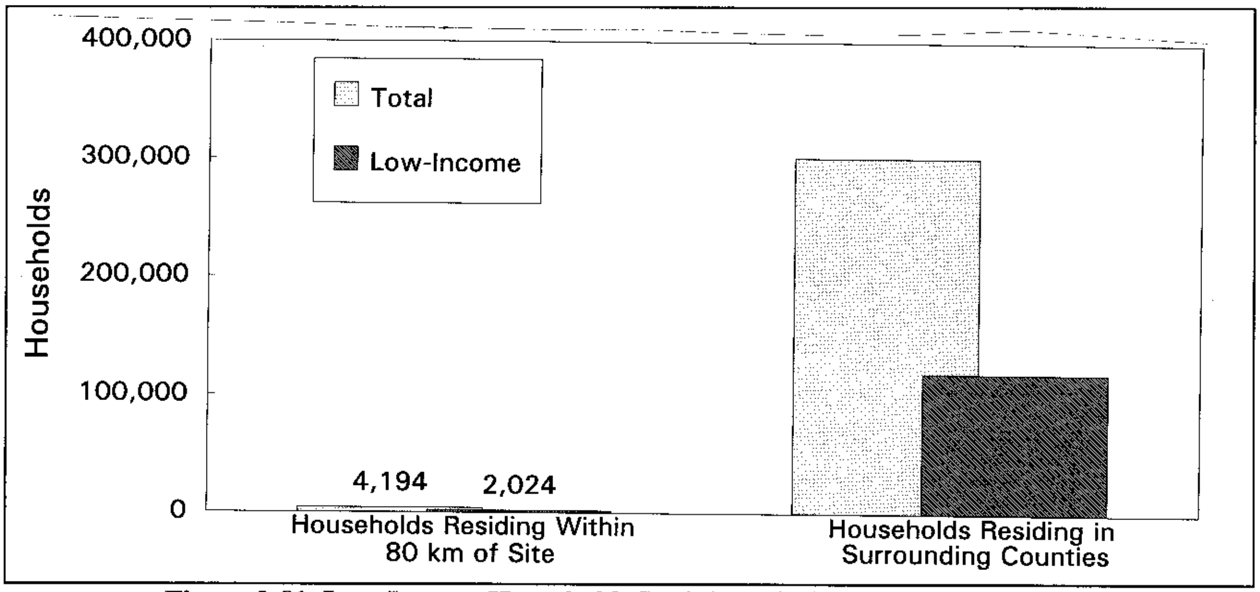


Figure 3-81 Low-Income Households Residing within 80 km (50 mi) of the Nevada Test Site

3.3.5.10 Historical, Archaeological, and Cultural Resources

People have inhabited the lands that comprise the Nevada Test Site for 12,000 years. The availability of the surface water was the primary determinant governing the location of past human occupation on these lands.

The Southern Paiute and Shoshone Native American tribes are known to have inhabited southern Nevada, including parts of what is now the Nevada Test Site. No known Native American resources are located on the Nevada Test Site (DOE, 1995c).

4. Policy Considerations and Environmental Impacts

This Chapter of the Environmental Impact Statement (EIS) describes the policy considerations and potential environmental impacts resulting from each of the management alternatives for implementation of the proposed action and the No Action Alternative. The environmental analysis addresses potential impacts of each alternative on workers, the public, and the environment. The general methodology used throughout this chapter is discussed in Section 4.1.

The policy considerations and environmental impacts of policy alternatives are described in this chapter. One policy alternative is the proposed action, which proposes the adoption of a policy whereby the United States would become involved in the management of the foreign research reactor spent nuclear fuel. The proposed action contains three separate management alternatives for adopting the policy. These management alternatives each contain different implementation alternatives related to that specific management alternative. The second policy alternative is the No Action Alternative which would involve no action by the United States in relation to the foreign research reactor spent nuclear fuel.

Each management alternative would result in very different policy considerations. Much of the foreign research reactor spent nuclear fuel analyzed in this EIS contains highly-enriched uranium (HEU), which can be used to make nuclear weapons. By adopting a policy to manage the foreign research reactor spent nuclear fuel, the proposed action would promote the U.S. goal of nuclear weapons nonproliferation by removing large amounts of HEU from civilian commerce. The No Action Alternative would be in direct conflict with the stated U.S. nuclear weapons nonproliferation goal and would seriously undermine credibility of the United States as a reliable partner in international nuclear weapons nonproliferation activities. Further, foreign research reactor operators may accuse the United States of failing to comply with its obligations under Article IV of the Non-Proliferation Treaty to share the benefits of peaceful nuclear cooperation with other countries.

Each management alternative would also result in very different environmental impacts in the United States which may vary according to the implementation alternatives of each management alternative. The No Action Alternative would have no direct environmental impacts in the United States.

Each of the three management alternatives under the proposed action is briefly summarized here. The three management alternatives were described in greater detail in Chapter 2, Sections 2.2 through 2.4. The policy considerations and environmental impacts of each alternative are described in detail in this chapter.

Management Alternative 1 — Manage Foreign Research Reactor Spent Nuclear Fuel in the United States

Management Alternative 1 of the proposed action entails acceptance and management of the foreign research reactor spent nuclear fuel in the United States. This management alternative would have direct environmental impacts in the United States.

Management Alternative 1 is composed of nine basic implementation components, as well as seven implementation alternatives that alter one of these basic components in some manner. The basic implementation of Management Alternative 1, as well as the seven implementation alternatives, are described in detail in Chapter 2, Section 2.2. The policy considerations and environmental impacts of the basic implementation of Management Alternative 1 are presented in Section 4.2. The policy considerations and environmental impacts of the seven implementation alternatives of Management Alternative 1 are presented in Section 4.3.

Management Alternative 2 — Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

Management Alternative 2 of the proposed action entails U.S. facilitation of overseas management of the foreign research reactor spent nuclear fuel at one or more foreign locations. No foreign research reactor spent nuclear fuel would be accepted into the United States. This would require advance negotiations and agreements with foreign reactor operators, officials in foreign governments, and reprocessing facilities. The outcome of these negotiations is uncertain. This management alternative would have no direct environmental impacts in the United States, unless the Department of Energy (DOE) decides to accept vitrified high-level waste from reprocessing facilities overseas in place of the foreign research reactor spent nuclear fuel. Very few countries have the capability to accept and store high-level wastes (GAO, 1994).

Management Alternative 2 is described in detail in Chapter 2, Section 2.3. Under this management alternative, the United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the foreign research reactor spent nuclear fuel containing U.S.-origin HEU. The policy considerations and environmental impacts of the two subalternatives of Management Alternative 2 are presented in Section 4.4.

Management Alternative 3 — Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

Management Alternative 3 entails some combination of the elements from Management Alternatives 1 and 2, and is referred to as the Hybrid Alternative. Management Alternative 3 would likely have more direct environmental impacts in the United States than Management Alternative 2, but less than Management Alternative 1.

Management Alternative 3 is described in detail in Chapter 2, Section 2.4. For purposes of analysis, a sample Hybrid Alternative has been included to demonstrate one possible combination of elements within Management Alternatives 1 and 2, and to allow an analysis of its impacts. It is important to note that the Hybrid Alternative described is merely an example for analysis purposes, and is only one of numerous possible combinations of elements from Management Alternatives 1 and 2.

Under the Hybrid Alternative described, DOE and the Department of State would facilitate the reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay, United Kingdom or Marcoule, France) for foreign research reactor operators in countries that can accept the reprocessing waste, as in Management Alternative 2. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States, as in Management Alternative 1. The policy considerations and environmental impacts of the sample Hybrid Alternative (Management Alternative 3) are described in Section 4.5.

Other Alternatives and Comparisons

The No Action and Preferred Alternatives are discussed in Sections 4.6 and 4.7, respectively. Comparisons across all the alternatives of the potential impacts and costs are presented in Section 4.8 and 4.9, respectively. Finally, this chapter concludes by comparing the risks due to the alternatives to the risks due to other common activities in Section 4.10.

4.1 Overview of Environmental Impacts

4.1.1 Presentation of the Environmental Impacts

Potential environmental impacts associated with each segment of the affected environment of the proposed action are addressed in this chapter. These segments are presented in this section in the following order:

- Marine transport impacts,
- Port of entry impacts,
- Ground transport impacts, and
- Management Site impacts.

The impact analyses of these four segments are described in more detail in Appendices C, D, E, and F, respectively. Effects of each implementation alternative of Management Alternative 1 of the proposed action on U.S. nuclear weapons nonproliferation goals and objectives are also discussed. In addition, this chapter summarizes the potential costs associated with the alternatives. Details on costs are presented in Appendix F.

Spent nuclear fuel is transported in strong, heavy casks (NRC, 1993). After the spent nuclear fuel is delivered, the empty casks must be transported back on a return trip. Under most of the alternatives, empty casks would be transported overland, through U.S. ports, and on ships. There would be minor nonradiological impacts (vehicle emissions and potential traffic accidents) during ground transport of empty casks. These nonradiological ground transport impacts are included as part of the assessment in this EIS.

4.1.2 Key Assessment Factors

A key assessment factor is one that may differentiate among alternatives, has a measurable impact, or be of public interest. The detailed analysis of potential environmental impacts presented in the appendices of this EIS did not reveal any factor likely to cause a large impact. Because radiation exposure and its consequences is a topic of great public interest, emphasis is placed upon exposure to radiation, although DOE considers the evaluated effects of radiation to be small.

During handling operations, the principal hazard would come from radiation being emitted by the foreign research reactor spent nuclear fuel. Without adequate shielding, the radiation levels at the surface of some of the spent nuclear fuel itself would often be high enough to induce a prompt fatality. This radiation can and would be attenuated (i.e., reduced) by the shielding materials of the transportation cask, such as lead, steel, and polyethylene. Further, since radiation intensity decreases with distance, maintaining a distance from the cask would also provide radiation protection. At 100 m (330 ft) from the cask, the radiation levels would not be detectable above background radiation. All foreign research reactor spent nuclear fuel handling at the proposed foreign research reactor spent nuclear fuel management sites would take place at considerable distances from the public (greater than 100 m or 330 ft). Recently, actual radiation measurements were taken by the State of North Carolina, Department of Environment, Health, and Natural Resources, of the casks used in the first shipment of the 153 spent fuel elements covered by the Urgent Relief Environmental Assessment (DOE, 1994m). In every case, the State of North Carolina reported detecting no radiation above background levels (radiation exposure from natural sources) at a distance of 1 meter (3.3 ft) from the package surface (State of North Carolina, 1994).

Accidents involving foreign research reactor spent nuclear fuel could potentially also result in releases of radioactive material which could cause radiation exposures. For most accidents, essentially none of the radioactive material would be released because it is an integral part of the solid fuel. Larger quantities of radioactive elements could be released only when the accident generates enough energy to release particles of foreign research reactor spent nuclear fuel to the atmosphere, such as with a fire. However, the probability of such accidents is very small. For most accidents, the energy would not be high enough to damage the foreign research reactor spent nuclear fuel, so that none of the radioactive material would be released.

4.1.3 General Radiological Health Effects

The effect of radiation on people depends upon the kind of radiation exposure (alpha and beta particles, and gamma and x-rays) and the total amount of tissue exposed to radiation. The amount of radiant energy imparted to tissue from exposure to ionizing radiation is referred to as absorbed dose. The sum of the absorbed dose to each tissue, when multiplied by certain quality and weighting factors that take into account radiation quality and different sensitivities of these various tissues, is referred to as effective dose equivalent (EDE).

An individual may be exposed to radiation from outside the body, or from inside the body because radioactive materials may enter the body by ingestion or inhalation. External dose is different from internal dose in that it is delivered only during the actual time of exposure. An internal dose, however, continues to be delivered as long as the radioactive source is in the body (although both radioactive decay and elimination of the radionuclide by ordinary metabolic processes decrease the dose rate with the passage of time). The dose from internal exposure is calculated over 50 years following the initial exposure.

The annual radiation dose limit to the public from nuclear facilities operated by DOE is 100 mrem per year (NRC, 1991). The potential foreign research reactor spent nuclear fuel management sites covered by DOE operations normally operate such that the public's dose is undetectable. For comparison, it is estimated that the average individual in the United States receives a dose of about 350 mrem per year from all sources combined, including natural and medical sources of radiation and radon. A modern chest x-ray, for example, results in an approximate dose of 8 mrem, while a diagnostic hip x-ray results in an approximate dose of 83 mrem (DOE, 1995c).

Radiation can also cause a variety of adverse health effects in people. A large dose of radiation can cause prompt death. At low doses of radiation, the most important adverse health effect for depicting the consequences of environmental and occupational radiation exposures (which are typically low doses) is the potential inducement of cancers that may lead to death in later years. This effect is referred to as latent cancer fatalities (LCF) because the cancer may take years to develop and for death to occur, and may never actually be the cause of death.

In addition to LCF, other health effects could result from environmental and occupational exposures to radiation. These effects include nonfatal cancers among the exposed population and genetic effects in subsequent generations. Table 4-1 shows the dose-to-effect factors for these potential effects as well as for LCF. For simplicity, this EIS presents estimated effects of radiation only in terms of LCF. The nonfatal cancers and genetic effects are less probable consequences of radiation exposure, and are less serious.

Table 4-1 Risk of LCF and Other Health Effects from Exposure to Radiation

Population[a]	*LCF*[b]	*Nonfatal Cancers*	*Genetic Effects*	*Total Detriment*
Workers	0.0004	0.00008	0.00008	0.00056
Public	0.0005	0.0001	0.00013	0.00073

[a] *The difference between the worker risk and the general public risk is attributable to the fact that the general population includes more individuals in sensitive age groups (that is, less than 18 years of age and more than 65 years of age).*

[b] *When applied to an individual, units are lifetime probability of LCF per rem of radiation dose. When applied to a population of individuals, units are excess number of cancers per person-rem of radiation dose. Genetic effects as used here apply to populations, not individuals.*

The collective or "population" dose to an exposed population is calculated by summing the estimated doses received by each member of the exposed population. This is referred to as a "population dose." The total population dose received by the exposed population is measured in person-rem. For example, if 1,000 people each received a dose of 0.001 rem, the population dose would be 1.0 person-rem (1,000 persons x 0.001 rem = 1.0 person-rem). The same population dose (1.0 person-rem) would result if 500 people each received a dose of 0.002 rem (500 persons x 0.002 rem = 1 person-rem).

The factor used in this EIS to relate a dose to its effect is 0.0004 LCF per person-rem for workers and 0.0005 LCF per person-rem for individuals among the general population (DOE, 1995c). The latter factor is slightly higher because of some individuals in the public, such as infants, who may be more sensitive to radiation than workers. These factors are based on the *1990 Recommendations of the International Commission on Radiological Protection* (ICRP, 1991), and are consistent with those used by the U.S. Nuclear Regulatory Commission (NRC) in its rulemaking *Standards for Protection Against Radiation* (NRC, 1991). The factors apply where the dose to an individual is less than 20 rem and the dose rate is less than 10 rem per hour. At doses greater than 20 rem, the factors used to relate radiation doses to LCF are doubled. At much higher doses, prompt effects, rather than LCF, may be the primary concern. Unusual accident situations that may result in high radiation doses to individuals are considered special cases. No such cases are expected with either incident-free handling or accidents with foreign research reactor spent nuclear fuel.

These concepts may be applied to estimate the effects of exposing a population to radiation. For example, if 100,000 people were each exposed only to background radiation (0.3 rem per year), 15 LCF per year would be expected (100,000 persons x 0.3 rem per year x 0.0005 LCF per person-rem = 15 LCF per year).

Sometimes, calculations of the number of LCF associated with radiation exposure do not yield whole numbers and, especially in environmental applications, may yield numbers less than 1.0. For example, if 100,000 people were each exposed to a total dose of only 1 mrem (0.001 rem), the population dose would be 100 person-rem, and the corresponding estimated number of LCF would be 0.05 (100,000 persons x 0.001 rem x 0.0005 LCF per person-rem = 0.05 LCF).

The *average* number of deaths that would result if the same exposure situation were applied to many different groups of 100,000 people is 0.05. In most groups, nobody (zero people) would incur an LCF from the one mrem dose each member would have received. In a small fraction of the groups, one latent fatal cancer would result; in exceptionally few groups, two or more latent fatal cancers would occur. The average number of deaths over all the groups would be 0.05 latent fatal cancers (just as the average of 0, 0, 0, and 1 is 1/4, or 0.25). The most likely outcome is zero LCF.

These same concepts apply to estimating the effects of radiation exposure on a single individual. Consider the effects, for example, of exposure to background radiation over a lifetime. The "number of LCF" corresponding to a single individual's exposure to 0.3 rem per year over a (presumed) 72-year lifetime is:

> 1 person x 0.3 rem per year x 72 years x 0.0005 LCF per person-rem = 0.011 LCF or one chance in 91 of an LCF.

Again, this should be interpreted in a statistical sense; that is, the estimated effect of background radiation exposure on the exposed individual would produce a 1.1 percent chance that the individual would incur a latent fatal cancer. Alternatively, this method estimates that about 1 person in 91 would die of cancers induced by background radiation.

4.1.4 Risks

Another concept important to the presentation of results in this EIS is the concept of risk. Risks are most important when presenting accident analysis results. The chance that an accident might occur during the conduct of an operation is called the probability of occurrence. An event that is certain to occur has a probability of 1.0 (as in 100 percent certainty). If an accident is expected to happen once every 50 years, the frequency of occurrence is 0.02 per year (1 occurrence every 50 years = 0.02 occurrences per year). A frequency estimate can be converted to a probability statement. If the frequency of an accident is 0.02 per year, the probability of the accident occurring in a 10-year program is 0.2 (10 years x 0.02 occurrences per year).

Once the frequency (occurrences per year) and the consequences (for radiation effects, measured in terms of the number of LCF caused by the radiation exposure) of an accident are known, the risk can be determined. The risk per year is the product of the annual frequency of occurrence times the number of LCF. This annual risk expresses the expected number of LCF per year, taking account of both the annual chance that an accident might occur and the estimated consequences if it does occur.

For example, if the frequency of an accident were 0.2 occurrences per year and the number of LCF resulting from the accident were 0.05, the risk would be 0.01 LCF per year (0.2 occurrences per year x 0.05 LCF per occurrence = 0.01 LCF per year). Another way to express this risk (0.01 LCF per year) is to note that if the operation subject to the accident continued for 100 years, one LCF would be likely to occur because of accidents during that period. This is equivalent to 1 chance in 100 that a single LCF would be caused by the accident source for each year of operation. This risk can be related to the risk of death from other accidental causes for comparison. As an example, the risk of dying from a motor vehicle accident is about 1 chance in 80. Similarly, the risk of death for the average American from fire is approximately 1 chance in 500, and for death from accidental poisoning, the risk is about 1 chance in 1,000 (NNPP, 1993). Section 4.10 compares the risks calculated in this EIS to those of common activities.

4.1.5 Estimated Radiation Dose Rate Near the Foreign Research Reactor Spent Nuclear Fuel Transportation Casks

The regulatory external radiation dose rate limit for foreign research reactor spent nuclear fuel transportation casks selected for use in the marine and ground transport analysis is 10 mrem per hour at 2 m (6.6 ft) from the "exclusive use" vehicle (no other cargo) [49 Code of Federal Regulations (CFR) 173.441]. This is equivalent to approximately 23 mrem per hour at 1 m (3.3 ft). Historical data from actual cask shipments of research reactor spent nuclear fuel have shown dose rates considerably below this regulatory limit. Dose measurements of casks containing research reactor spent nuclear fuel, including the foreign research reactor spent nuclear fuel recently received under the Urgent Relief Environmental

Assessment (DOE, 1994m), are presented in Appendix F, Section F.5. The average of these measurements is 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the cask. Recent measurements taken by the State of North Carolina on foreign research reactor spent nuclear fuel shipment packages, covered by the Urgent Relief Environmental Assessment, showed that the external dose rate at 1 m (3.3 ft) was undetectable above background radiation levels (State of North Carolina, 1994).

To be conservative, the analyses in this chapter use the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the side of the transport vehicle for the radiation dose rate near the foreign research reactor spent nuclear fuel casks. This conservative value was used in the calculations of incident-free doses to members of the public, marine transport workers, port workers, and ground transport workers. For radiation workers at the spent nuclear fuel management sites, the dose rate in the vicinity of the casks was estimated by the conservative methodology presented in Appendix F, Section F.5.

4.1.6 The Effects of Radiation on Plants and Animals

There is no convincing evidence from the scientific literature that chronic radiation doses below 1 rad per day will harm animal or plant populations. It is highly probable that limitation of the exposure of the most exposed humans (the critical human group, living on and receiving full sustenance from the local area) to 100 mrem per year will lead to dose rates to plants and animals in the same area of less than 1 rad per day. DOE and NRC regulations limit annual human exposures to values far lower than those that have caused observable damage in plant and animal populations. Therefore, specific radiation protection standards for nonhuman biota are not needed (IAEA, 1992).

4.2 Management Alternative 1 – Manage Foreign Research Reactor Spent Nuclear Fuel in the United States – Basic Implementation

This section presents the policy considerations and potential environmental impacts of the basic implementation of Management Alternative 1. Under the basic implementation of Management Alternative 1, all the foreign research reactor spent nuclear fuel could be accepted into the United States. DOE and the Department of State believe this would promote the nuclear weapons nonproliferation objective of reducing, and ultimately eliminating, civil commerce in HEU. The spent nuclear fuel could be managed safely and securely at any of five DOE sites.

Policy Considerations

A critical result of this basic implementation of Management Alternative 1 would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs. The successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States commitment to action. Finally, this basic implementation of Management Alternative 1 would support the Administration's nuclear weapons nonproliferation objective of not encouraging reprocessing for either nuclear power or nuclear explosive purposes.

Another crucial consideration associated with Management Alternative 1 is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other

Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

The amount of foreign research reactor spent nuclear fuel that would be accepted under the basic implementation of Management Alternative 1 is up to approximately 19.2 metric tons of heavy metal (MTHM) representing approximately 22,700 elements. This amount is an upper limit because if some nations were to reprocess their research reactor spent nuclear fuel, for example, the amount of foreign research reactor spent nuclear fuel accepted into the United States would be reduced. Under the basic implementation of Management Alternative 1, approximately 4.6 metric tons (5.1 tons) of HEU would be removed from international commerce.

4.2.1 Marine Transport Impacts

Because the basic implementation of Management Alternative 1 involves ocean transport, DOE and the Department of State considered the environmental impacts on the global commons (i.e., portions of the ocean not within the territorial boundary of any nation) in accordance with Executive Order 12114 (U.S. Federal Register, 1979).

4.2.1.1 General Assumptions and Analytic Approach

The basic implementation of Management Alternative 1 includes the shipment of approximately 837 transportation casks containing foreign research reactor spent nuclear fuel over a 13-year period. Of these, approximately 721 transportation casks would be transported by sea to the United States, with the remainder (116) coming overland from Canada. DOE would prefer to consolidate the approximately 721 casks on board ships to minimize the number of voyages, but it is also possible that approximately 721 voyages could be required. This section evaluates the impacts of the marine transportation, including shipment in international waters from the port of origin to the United States and coastal shipping in United States territorial waters.

Four types of commercial cargo ships are considered to be candidates to carry foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1: containerized, breakbulk (general cargo), roll-on/roll-off, and purpose-built vessels (see Appendix C for a more complete description of these vessels). DOE and the Department of State assumed that all casks would be transported in standard International Standards Organization 20-ft shipping containers, because this is current shipping practice.

Nonradiological impacts associated with the marine shipment of 721 containerized transportation casks would be minimal. The United States receives more than 56,000 ships engaged in foreign trade at its ports each year (DOC, 1994). Shipping an additional 56 containers per year on average over the 13-year receipt period is not likely to cause any additional ships to sail beyond the number already scheduled. In the event that chartered vessels are used for this program, up to 10 voyages per year could be required, which is only 0.02 percent of the number engaged in regular commerce. Additional nonradiological impacts would be very small whether chartered or regularly scheduled commercial vessels are used. The number of containers handled on a regular basis is so large that the addition of the foreign research reactor spent nuclear fuel containers would add essentially no impacts (cargo vessels typically carry 800 to 1,000 containers per voyage). While nonradiological marine events such as unloading or cargo shifting accidents would be possible, the nonradiological impacts would be miniscule.

The radiological impacts of transporting the foreign research reactor spent nuclear fuel by sea were considered in two ways, incident-free impacts and accident impacts. The incident-free impacts would be those that occur simply due to the marine shipping of foreign research reactor spent nuclear fuel, assuming there are no accidents. The ship's crew would be the affected individuals in this case. The accident impacts would be the consequences of reasonably foreseeable accidents that might occur. These two evaluations are discussed in the following two sections, with additional details in Appendix C.

4.2.1.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Marine Transport

The primary impact of incident-free marine shipping of foreign research reactor spent nuclear fuel would be upon the crews of the ships used to carry the spent nuclear fuel casks. Since the crew of a ship is normally separated from the cargo and shielded by both the cargo and the ship's structure, the risk to the crew from spent nuclear fuel transport during most crew activities would be extremely low (DOE, 1994m). The exceptions would include the exposure to the crew during loading and off-loading of the spent nuclear fuel ISO containers and during daily inspection of the ship's cargo, including the containers housing the spent nuclear fuel transportation casks. Therefore, the crew exposure during loading, daily inspection, and unloading of the transportation casks has been incorporated into the incident-free marine transport analysis. The exposure to dock workers at the foreign research reactor spent nuclear fuel port of entry is assessed in Section 4.2.2.

Daily inspections of the casks is the activity that would result in the largest doses to the ship's crew, with the inspectors considered the maximally exposed workers during incident-free marine transport. For any given voyage, DOE and the Department of State conservatively assumed that the same three inspectors would conduct all of the inspections. The impact on the inspectors would be a function of the number of inspections performed, which would depend upon the amount of time the cask is onboard. Therefore, the incident-free radiological impact on the inspectors would depend upon the total duration of the voyage, including days at sea, in intermediate ports, and days in coastal sailing between intermediate ports. The duration of the voyage was selected as the weighted average of the duration of all the shipments necessary for 721 transportation casks. (See Appendix C for further details regarding this assumption.)

To maximize the estimated impact from incident-free transport, DOE and the Department of State made conservative assumptions regarding crew exposure. Specifically, DOE and the Department of State conservatively assumed that eight and two casks (loaded two casks per hold) would be shipped per voyage of chartered and regularly scheduled commercial ships, respectively. This assumption would result in additional exposure of the ship's crew due to the effect of loading casks into holds where a loaded cask would have already been stowed, and would also increase the exposure to the crew members performing daily inspections. The additional exposure would be a result of the combination of the radiation fields surrounding each of the transportation casks.

Assuming 56 casks per year, the number of annual voyages required would range from 7 to 28, depending upon the number of casks per ship. Although the foreign research reactor spent nuclear fuel would be shipped from 40 countries worldwide and to both U.S. coasts over a 13-year receipt period, DOE and the Department of State conservatively assumed that a single crew could be involved in up to 9 voyages per year. As a practical matter, this overstates the rate at which a crew would sail from Europe or Asia and back. Additionally, to determine the dose to the maximally exposed worker in the ship's crew, DOE and the Department of State conservatively assumed that the same individuals would conduct all the daily onboard inspections.

The dose received during daily cargo inspection would be a major contributor to the crew dose, so the duration of the voyage is an important consideration. Chartered vessels would sail directly to the port(s) of entry, yielding an average voyage duration of 18 days. DOE and the Department of State conservatively assumed that all shipments aboard regularly scheduled commercial breakbulk vessels would include two intermediate port stops in the United States, which would add 3 days to the voyage.

Table 4-2 presents the maximum estimated incident-free marine transport doses and risks. Values are provided for a chartered ship (which would not make intermediate port calls) and for a regularly scheduled commercial vessel. The values are based on the estimated time the cask would be onboard multiplied by the dose per day received as a result of inspections, plus the crew dose due to the foreign research reactor spent nuclear fuel container loading and off-loading activities. While the use of a chartered ship would result in higher per-shipment impacts (eight casks per shipment versus two for regularly scheduled commercial ships), the reduced number of voyages would offset this increase in per-shipment impacts. Therefore, the use of chartered ships instead of regularly scheduled commercial ships would result in slightly lower total crew exposures in the basic implementation of Management Alternative 1. The selection of the shipping mode, however, would not be based on crew exposures alone. Other factors, such as cost, would also be important in the choice of chartered or regularly scheduled ships. The results in Table 4-2, therefore, provide an estimate of the range of maximum worker exposures due to the shipment of the foreign research reactor spent nuclear fuel.

Table 4-2 Incident-Free Marine Transport Impacts[a]

	Regularly Scheduled Commercial Ship				*Chartered Ship*			
	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Risk (LCF)*	*Maximally Exposed Worker Dose (mrem)*	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Risk (LCF)*
Impacts Per Shipment	66[b]	0.000027	0.23	0.000091	100[b]	0.00004	0.83	0.00033
Impacts for the Basic Implementation	1,300[b,c]	0.00052[c]	85	0.034	1,300[b,c]	0.00052[c]	75	0.030

[a] *These results are based on the assumption that the dose rates associated with the casks are all derived from the exclusive-use regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of this regulatory limit, so this assumption is conservative.*

[b] *If an individual works on repeated shipments, this maximally exposed worker dose could exceed the annual regulatory limit. Therefore, DOE would require that mitigation measures be implemented to keep the maximally exposed worker dose down to 100 mrem per year or lower. See Appendix C for estimates of the total exposure to the ships' crews without mitigation measures.*

[c] *These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years.*

Marine transport workers are not trained to be radiation workers, so they would not be subject to the radiation worker limit of 5,000 mrem/yr. The applicable regulatory limit for these workers would be the same as for the general public: 100 mrem/yr. As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest estimated population risk is about 0.034 LCF, which is much less than 1 LCF.

4.2.1.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Marine Transport

The basic implementation of Management Alternative 1 could potentially impact the marine environment in the event of an accident involving the release of radioactive material from the spent nuclear fuel. This section discusses possible accidents and their consequences.

The range of accidents that could occur during marine transport is quite broad. The ship could collide with another vessel or an object such as a shoal, rock, or wreck. Foul weather could damage or sink the ship, or the ship could experience a fire, explosion, or other problem. To reduce the risk due to potential accidents, the casks that would carry the foreign research reactor spent nuclear fuel have been designed to prevent damage to the cask contents in all but the most severe, and least likely, cases. See Appendix B of this EIS for a description of the foreign research reactor spent nuclear fuel transportation casks.

Two scenarios emerge that could potentially threaten the marine environment and possibly humans: the cask could be damaged and then involved in a fire, or the cask could sink. These cases are discussed in more detail below.

Cask Damaged Followed by a Fire

A ship carrying foreign research reactor spent nuclear fuel could be involved in a severe collision with another ship. It is possible that a transportation cask, carried on a ship involved in such a collision, could be exposed to impact forces resulting from the collision. In that event, the cask could be damaged. However, only a small fraction, at most, of the force generated in a collision of one ship with another would be brought to bear on a transportation cask for two reasons. First, the force of a ship-to-ship collision would be distributed over the entire area of contact between the two ships, which means that the force density (force per unit of area) that would result from a collision must be considered. The maximum cross sectional area presented by a transportation cask would be small in comparison to the typical impacted area, so that even if a cask were located directly in the path of the collision and unprotected by intervening hulls, bulkheads, etc., the force that might be exerted on such a cask would be limited by the force density.

Secondly, ships floating on water are yielding objects, so that some portion of the energy of impact would be transmitted to the water. Even severe collisions with large impact forces, by themselves, would not necessarily result in catastrophic failure of a transportation cask. Thus, it would be even more unlikely for a less severe collision to result in the breach of a cask and, thereafter, a release of any of its contents. Attachment 4 to Appendix D discusses in detail the forces involved in ship collisions.

If it is assumed, however, that a ship collision breaches the cask, a release of radioactive material would be possible. In such a circumstance, the release would be small because the spent fuel is metallic and thus would release very limited quantities of radioactive material, even if mechanically damaged. However, due to the severity of the collision required to breach the cask, the ship carrying the foreign research reactor spent nuclear fuel cask would be severely damaged and probably would sink. Whether the ship would sink or not, the only humans that could be affected (the crew) would most likely not be in the vicinity of the impact point of the collision, where the damaged cask would be located.

The limiting accident is a ship collision severe enough to breach a cask carrying foreign research reactor spent nuclear fuel and also cause a large fire. Some of the radioactive contents of the cask could be released and carried into the air by the heated gases of the fire as a plume of radioactive particles. For an airborne release of this type to occur, the cask-carrying vessel must stay afloat during and immediately after the accident. In practice, this would mean that the ship must stay afloat for a period of some hours following an accident of the requisite severity. This latter condition must be satisfied for atmospheric dispersal to occur, even though marine casualty files indicate that a common outcome of severe ship collisions is rapid sinking, often within a matter of minutes. Assuming the cask was damaged by a severe collision; and the ship remained afloat despite the severe collision; and the cask was engulfed in flames for a time sufficient to release a radioactive plume, there would likely be no human population on the ocean (excluding the crew) who could be affected.

It is possible that the ship could be in coastal waters (i.e., beyond the port's sea buoy) at the time of this severe collision. Except in port, a ship is seldom within 16 km (10 mi) of a population center, so the port accident public risk analysis in the next section covers public risk in this scenario. The ship's crew and people onboard other vessels that may come to provide assistance could be exposed to any released radioactive material. The number of people potentially exposed would be less than that used in the port accident analysis for populations near a port [less than 1.6 km (1 mi) from the port]. Additionally, accident frequencies at sea tend to be lower than in-port accident frequencies. Therefore, both the consequences and risks for an accident at sea are covered by the results of the port accident analysis.

Risks associated with this type of accident at sea are covered by the risks of the same type of accident in ports because humans in the vicinity of the accident at sea are much fewer in number than even the least populated port.

Sunken Cask

The second scenario of concern is that a foreign research reactor spent nuclear fuel cask or casks would be sunk. This could be the result of the ship sinking, of the casks being somehow swept overboard, or of a ground transport accident on a causeway. Submersion of an intact cask would not necessarily result in a release of its contents, as spent nuclear fuel casks are designed to withstand at least a 15 m (50 ft) immersion. It has been demonstrated that cask seals will remain intact at much greater depths (DOE, 1994m). Should a loaded foreign research reactor spent nuclear fuel cask (damaged or undamaged) sink anywhere in the U.S. coastal waters, it will be recovered regardless of depth. U.S. Coastal waters in this case refers to waters within the 12 mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, spent nuclear fuel casks can, and likely would, be recovered from water up to 200 m (660 ft) deep, which is beyond the range typical of coastal and port depths. Typically 200 m (660 ft) is considered the limit of the continental shelf. Recovery at depths greater than 200 m (660 ft) is possible but is more difficult.

If a sunken cask containing foreign research reactor spent nuclear fuel were recovered, the effect on the marine environment would be minimal, even if the recovery effort required up to 1 year to complete. The release to the ocean water of radioactive particles from the spent nuclear fuel requires that first the metallic spent fuel corrode, then the radioactive particles escape from the cask. Even if the cask were damaged, the most likely damage to a spent nuclear fuel cask, either from mechanical trauma or excessive depth, would be failure of the seal. Seal failure would allow seawater to enter the cask to begin the corrosion of the metallic spent nuclear fuel, but the flow of water through the cask to carry out the radioactive material would be minimal due to the small cross sectional area of the failed seal. The decay heat from the spent nuclear fuel is low, thereby providing no driving force to expel water out of the cask through the failed seal.

If a cask was not recovered, the radioactive constituents of spent nuclear fuel would be released slowly over time into the surrounding waters. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a cask were submerged on the deep ocean bottom and not recovered, the peak human dose to an individual ingesting seafood harvested from the area in which the breached submerged spent nuclear fuel cask would be located would be 114 mrem per year. If a sunken cask in coastal waters was not recovered, the peak human dose is conservatively estimated to be 14,000 mrem per year. Consequences to humans and to marine biota are presented in Table 4-3. Other studies of similar circumstances indicate that the individual dose would be even lower (DOE, 1980). Uranium (the major constituent of the spent nuclear fuel) has been found not to bioaccumulate in fish, and bioaccumulates only slightly in crustaceans and mollusks (IAEA, 1976). The peak doses for humans, fish, crustaceans, and mollusks are presented in Table 4-3 in the situation where a chartered ship carrying eight casks might sink in deep ocean. Doses for humans and other animals are expressed in units of rem and rad, respectively. Rem is discussed in some detail in Section 4.1.3. While rem is only used for measuring human exposure to radiation, rad is used to measure exposure of nonhumans to radiation. Rad is a unit of absorbed dose from ionizing radiation.

The probability provided in Table 4-3 is the probability of one ship accident and loss of a cask during the entire program. The consequences are from one unrecovered cask. The program risk is the product of the probability and the consequences. Humans would not be the principally exposed species in a marine accident involving foreign research reactor spent nuclear fuel. Estimates were made of the dose to the biota received from a damaged cask containing foreign research reactor spent nuclear fuel. This analysis assumes that the cask would lay on the deep ocean floor where it would slowly release its radioactive inventory whether it was damaged in the collision or not.

Table 4-3 Impacts of Unrecovered Casks in Deep Ocean

	Probability	*Consequences*	*Program Risk*
MEI (human)	1.7×10^{-6}	114 mrem/yr	0.00019 mrem/yr
Fish	1.7×10^{-6}	640 rad/yr	1.1 mrad/yr
Crustaceans	1.7×10^{-6}	880 rad/yr	1.4 mrad/yr
Mollusks	1.7×10^{-6}	30,000 rad/yr	49 mrad/yr

Risks associated with the release of the contents of the spent nuclear fuel elements into the deep ocean are expected to be very small due to the low probabilities and limited consequences. The highest estimated risk to the MEI is 0.00019 mrem per year for every year that the cask leaks and this hypothetical individual ingests seafood harvested from near the cask. DOE and the Department of State assume that these conditions could apply for about 5 years, so the total MEI dose would be 0.00095 mrem. This translates into a maximum estimated MEI risk of 5×10^{-10} LCF. This means that this hypothetical individual's additional chance of incurring an LCF would be less than one in a billion. The risks to fish, crustaceans, and mollusks are low enough that no adverse impacts would be expected.

Probabilities, consequences, and risks were also calculated for the cases of unrecovered casks in coastal waters, both undamaged and damaged. The results are presented in Table 4-4, again in terms of rem for humans and rad for other animals. In coastal waters, cask recovery is considered likely (NEA, 1988), which makes the probabilities in Table 4-4 low. Comparing Tables 4-3 and 4-4 shows that the consequences of a sunken cask in coastal waters would be greater than in the deep ocean, but when multiplied by the probabilities, the risks are actually lower.

Table 4-4 Impacts of Unrecovered Casks in Coastal Waters

	Probability One Undamaged Cask	*Consequences One Undamaged Cask*	*Program Risk*
MEI (human)	2.3×10^{-8}	190 mrem/yr	4.3×10^{-6} mrem/yr
Fish	2.3×10^{-8}	77 mrad/yr	1.8×10^{-6} mrad/yr
Crustaceans	2.3×10^{-8}	81 mrad/yr	1.9×10^{-6} mrad/yr
Mollusks	2.3×10^{-8}	210 mrad/yr	4.8×10^{-6} mrad/yr
	Probability One Damaged Cask	*Consequences One Damaged Cask*	*Program Risk*
MEI (human)	4.6×10^{-11}	14,000 mrem/yr	6.4×10^{-7} mrem/yr
Fish	4.6×10^{-11}	620 mrad/yr	2.9×10^{-8} mrad/yr
Crustaceans	4.6×10^{-11}	660 mrad/yr	3.0×10^{-8} mrad/yr
Mollusks	4.6×10^{-11}	14,000 mrad/yr	6.4×10^{-7} mrad/yr

These risk estimates were derived assuming that the foreign research reactor spent nuclear fuel is shipped at a rate of one cask per voyage. Assuming a different shipping schedule, such as eight casks per voyage, would not result in a different estimate of the risks. The potentially higher consequences of an accident involving more than one shipping cask would be balanced by the reduced probability of an accident due to the reduced number of shipments. For example, the risk associated with one shipment of eight casks is equivalent to the risks associated with eight single cask shipments.

4.2.1.4 Marine Transport Cumulative Impacts

The cumulative impact of radioactive material shipments on ships' crews beyond that discussed in Section 4.2.1.2 was not estimated. In estimating the cumulative impact on port workers (see the following section) it was possible to estimate the total number of shipments of radioactive material through a port. However, it is not as simple to estimate the total number of shipments of radioactive material that involve the same ship and crew. It is expected that each ship's crew would be exposed to fewer of the shipments of radioactive material than that assumed for the port worker in the cumulative impact analysis for the port. For port workers, the impacts of the shipments other than the foreign research reactor spent nuclear fuel were of the same order of magnitude, but lower than the foreign research reactor spent nuclear fuel shipments. Therefore, the individual crew member's exposure from shipments other than the foreign research reactor spent nuclear fuel shipments would be a small fraction of the dose received due to the foreign research reactor spent nuclear fuel shipments.

4.2.1.5 Marine Transport Mitigation Measures

The principal environmental impact that would occur during marine transport would be radiation dose to the ships' crews. Most of this dose occurs because crew members must visually inspect the cargo every day for safety reasons, and the inspections cannot be curtailed.

The magnitude of the estimated impacts from this portion of the basic implementation of Management Alternative 1 is primarily due to two items: the conservative assumption that the radiation field emanating from all of the casks would be at the regulatory limit (as opposed to the levels of one-tenth of the regulatory limit that have been observed in past foreign research reactor spent nuclear fuel shipments), and the conservative assumption that the same crew member is involved in inspections for all of the casks on nine shipments during any given year. In reality, neither of these conservative assumptions would be likely to occur. Nevertheless, to ensure that no member of a ship's crew could receive a dose above what is allowed for a member of the general public, DOE would mitigate this effect by implementing a system through its shipping contractor to track each ship and crew involved in the shipment of foreign research reactor spent nuclear fuel. DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring that other crew members be used if any crew member approaches a 100 mrem dose in any year.

If a cask or casks were sunk in deep ocean or coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2 Port Activities Impacts

4.2.2.1 General Assumptions and Analytic Approach

To assess the range of potential impacts on ports at which a ship carrying foreign research reactor spent nuclear fuel might call, 13 ports of entry representing a wide range of port city population densities were selected for detailed evaluation. Eight of the ports—Charleston, SC; Elizabeth, NJ (for the New York City area); Philadelphia, PA; Norfolk, VA (representing Hampton Roads); Jacksonville, FL; Savannah, GA; Wilmington, NC; and Military Ocean Terminal at Sunny Point (MOTSU), NC—are East Coast ports that represent high, medium, and low population density ports. The Norfolk Terminal was selected to represent the three terminals (Newport News, Norfolk, and Portsmouth) at Hampton Roads for the analysis of potential impacts because this terminal provides the most conservative values in terms of estimated impacts. The West Coast ports chosen for evaluation were Long Beach, CA; Concord Naval Weapons Station (NWS), CA; Portland, OR; and Tacoma, WA, to represent high and medium population density ports. On the Gulf Coast, Galveston, TX was analyzed. These ports were selected to represent a range of ports in this analysis, not necessarily as the chosen ports of entry for foreign research reactor spent nuclear fuel. Ports representative of a group of ports with similar characteristics (i.e., of similar population around the port) were selected for analysis rather than attempting to analyze accidents at every potential port. Actual port selection and specific selection criteria are discussed in Appendix D, Section D.1.

The analysis assumed that there were no restrictions on the shipping routes taken by the cargo vessel carrying the foreign research reactor spent nuclear fuel. This assumption allows the vessel to make intermediate stops at any U.S. port capable of unloading the vessel. This implies that the vessel could enter most ports capable of receiving ocean-going cargo vessels, a group of ports that far outnumbers the ports that survive the port selection criteria for the receipt of foreign research reactor spent nuclear fuel. It was conservatively assumed that regularly scheduled commercial ships carrying foreign research reactor spent nuclear fuel would pass through two intermediate U.S. ports before reaching the port of entry for the foreign research reactor spent nuclear fuel. The 13 ports with high, medium, and low population densities that were chosen for site-specific accident analysis provide a perspective on the accident risks at the more than 100 ports that could be intermediate ports of call for the foreign research reactor spent nuclear fuel vessels.

Each port stop would or could involve:

- Port entry from the sea buoy,
- Docking,
- Inspection of cargo,

- Partial unloading of cargo,
- Partial reloading of cargo, and
- Port exit to the sea buoy.

As with the marine transport, the port impacts were evaluated for two conditions: incident-free and accident conditions. Summary results are presented in the following sections. Details of the analysis are presented in Appendix D.

4.2.2.2 Conservative Assumptions and Maximum Estimated Impacts of Incident-Free Port Activities

As stated in Section 2.6, no spent nuclear fuel transportation cask has ever released its contents (radioactive material), even as a result of an accident. For this reason, release of radioactive material is not considered as part of the incident-free analysis. The only impact considered is that caused by radiation exposure due to radiation emitted by foreign research reactor spent nuclear fuel contained within the transportation casks. Since no radioactive material would be released, there would be no impacts on land, water, or air quality in any of the ports or any of the waterways used by ships in the transport of foreign research reactor spent nuclear fuel.

Risks associated with the foreign research reactor spent nuclear fuel in incident-free conditions in port are predominantly those to inspectors and port workers. Port workers and inspectors are not radiation workers as defined by NRC regulations. Thus, the maximum allowable annual exposure for these personnel would be 100 mrem, the same radiation dose limit established by the NRC to protect individual members of the public (DOE, 1990c). When a ship arrives in its first port, the spent nuclear fuel package would be inspected by customs officials, U.S. Coast Guard personnel, and others. Up to six inspections, estimated at up to 15 minutes per person per spent nuclear fuel cask, were conservatively assumed. Once inspections are complete, the ship would partially unload and reload cargo. After that, DOE and the Department of State conservatively assumed that the ship would proceed to another intermediate port and then to the port of entry for the foreign research reactor spent nuclear fuel.

To determine the incident-free risks associated with port operations, two types of ships were considered for the shipment of the foreign research reactor spent nuclear fuel. In the first case, DOE and the Department of State conservatively assumed that all shipments were made on regularly scheduled commercial breakbulk ships. This type of vessel was selected because it maximized the time required for port activities, such as off-loading and inspections. In addition, during the operations at the intermediate port stops, DOE and the Department of State conservatively assumed that other unloading and loading operations would occur in the vicinity of the container with the loaded foreign research reactor spent nuclear fuel cask in one of the intermediate ports. Risks associated with these activities, which are comparable to the risks associated with the off-loading of the foreign research reactor spent nuclear fuel, have been included in the assessment. Transport of the material on this type of vessel would therefore result in the highest worker radiation doses in the incident-free analysis. All worker exposures were calculated by estimating the times required for activities and the distances from the transportation cask to where the worker would most likely be located.

To provide a measure of the difference in the worker exposures resulting from the use of cargo vessels other than the regularly scheduled commercial breakbulk vessels, the analysis was also performed for port operations associated with the use of a chartered container vessel. This type of vessel requires the least amount of time to unload. DOE and the Department of State also assumed that a chartered vessel would not make any intermediate port stops, so that the ship's port of entry into the United States would also be the port of entry for the foreign research reactor spent nuclear fuel. Use of these two types of vessels in the analysis provides an estimate of the range of the maximum incident-free risk associated with port operations.

At the port of entry, the casks would be off-loaded by port workers, and arrangements would be made for the immediate departure of the foreign research reactor spent nuclear fuel from the port. In recognition of instances where some delay may occur, DOE and the Department of State conservatively assumed a delay of up to 24 hours in a secure staging area. The 24-hour period for the staging of spent nuclear fuel casks was selected because it is possible that, on occasion, the spent nuclear fuel casks would not leave the secure staging area the same day that they arrived, depending on variables such as the time of day the casks clear customs and the weather. Nonetheless, DOE and the Department of State consider it unlikely that the casks would remain in the staging area for longer than 24 hours.

To estimate the maximum individual exposure, the shipments were divided into East Coast and West Coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South America were designated as East Coast shipments, all others were designated as West Coast shipments. Under these assumptions, the East Coast port(s) would receive approximately 535 casks and the West Coast port(s) approximately 186 casks. DOE and the Department of State also conservatively assumed for this analysis that all the shipments would pass through the same intermediate ports as the shipments on regularly scheduled commercial vessels and have the same port of entry.

Further, DOE and the Department of State made the very conservative assumption that the same inspectors and workers would handle every cask shipment. The per-shipment doses were then multiplied by the number of shipments for the East Coast to determine the maximally exposed worker dose for the basic implementation of Management Alternative 1.

In determining the worker population exposure, all shipments (East Coast and West Coast) were considered. This results in the integrated dose for the entire basic implementation of Management Alternative 1 which would span 13 years. The maximum estimated incident-free risks to port personnel due to the basic implementation of Management Alternative 1 are presented in Table 4-5. The incident-free risk to the general public would be zero because only workers would be near the casks in port.

This table shows the maximally exposed worker dose, worker population dose, and associated risks for the shipment of foreign research reactor spent nuclear fuel as containerized cargo on a regularly scheduled commercial breakbulk vessel and as cargo on a chartered container vessel. These figures represent the range of maximum estimated impacts for the various shipping modes available for the ocean transport of foreign research reactor spent nuclear fuel.

As the table shows, the highest estimated maximally exposed worker risk is 0.00052 LCF, which is based on the annual regulatory limit every year for 13 years. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest total population risk for port workers is 0.012 LCF, which is much less than one LCF.

Table 4-5 Incident-Free Port Activity Impacts[a,b]

Impacts per Cask Transfer

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers per Cask (person-rem)	Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers per Cask (person-rem)	Population Risk (LCF)
Inspectors	3.8	0.0000015	0.013	0.0000052	1.3	5×10^{-7}	0.0053	0.0000021
Port Handlers, Intermediate Ports	2.2	9×10^{-7}	0.018	0.0000071	----	----	----	----
Port Handlers, Port of Destination	2.0	8×10^{-7}	0.0066	0.0000026	0.46	1.8×10^{-7}	0.0015	6×10^{-7}
Port Staging Personnel	0.36	1.4×10^{-7}	0.0045	0.0000018	0.4	2×10^{-7}	0.0046	0.0000018
Maximum	3.8	0.0000015	----	----	1.3	5×10^{-7}	----	----
Total	----	----	0.042	0.000017	----	----	0.011	0.0000045

Impacts for the Entire Basic Implementation

Risk Group	Regularly Scheduled Commercial Breakbulk Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers (person-rem)	Population Risk (LCF)	Chartered Container Ship: Maximally Exposed Worker Dose (mrem)	Maximally Exposed Worker Risk (LCF)	Population Dose to Workers (person-rem)	Population Risk (LCF)
Inspectors	1,300[c]	0.00052[c]	9.4	0.0038	670	0.00027	3.8	0.0015
Port Handlers, Intermediate Ports	1,186	0.00047	13	0.0052	----	----	----	----
Port Handlers, Port of Destination	1,072	0.00043	4.8	0.0019	250	0.0001	1.1	0.00044
Port Staging Personnel	190	0.000076	3.2	0.0013	210	0.000084	3.3	0.0013
Maximum	1,300[c]	0.00052	----	----	670	0.00027	----	----
Total	----	----	30	0.012	----	----	8.2	0.0032

[a] *These results are based on the assumption that the dose rates associated with the casks are all based on the regulatory limit. Historically, the average of these dose rates has been equal to about one-tenth of the regulatory limit, so this assumption is conservative.*

[b] *These results are all based on the assumption that each voyage carries two casks. This assumption is conservative because chartered ships may carry up to eight casks.*

[c] *With all the conservative assumptions in this analysis, the maximally exposed worker dose could theoretically exceed the annual regulatory limit. Therefore, DOE would require mitigation measures to keep the maximally exposed worker dose down to 100 mrem per year or lower. These results are based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. See Appendix D for maximally exposed worker doses without mitigation measures.*

4.2.2.3 Conservative Assumptions and Maximum Estimated Impacts of Accidents During Port Activities

Section 4.2.1.3 discussed the impacts of marine accidents that could occur either in the open ocean or during coastal passages. This section discusses the impacts of accidents that could occur anywhere from the sea buoy into the port and at the pier.

Methodology

An analysis of reasonably foreseeable accidents must evaluate the consequences of possible accidents and the probability of an accident occurring. In incident-free marine transport, some exposure would be expected from radiation emitted from the casks. In the case of accidents, the probability of exposure is only an estimate of a hypothetical event. Accident probabilities are derived from published maritime accident rates. The analysis of ship collisions concludes that only one hold of the ship carrying the foreign research reactor spent nuclear fuel transportation casks would be subject to sufficient forces to potentially result in cask damage. There is no difference between the risks associated with a single shipment with two casks in a hold, and two shipments of a single cask each. The consequences of the accident with two casks in the hold may be as large as twice the consequences of an accident involving one cask. But the probability of an accident involving the ship carrying the two casks is half the probability of one of the two ships carrying a single cask being involved in an accident. Therefore, the potential risk from accidents, marine transportation of spent nuclear fuel was modeled in the port accident analysis as occurring in one cask per shipment.

Because accidents can be of any degree of severity, from a "fender bender" to one involving severe impact and prolonged fire, the severity spectrum is divided into a number of accident severity categories. Each category is assigned a conditional probability of occurrence [i.e., a probability (given that an accident occurs) that it will be of that particular severity]. In general, the more severe the accident, the more remote the chance of such an accident. In this analysis, the accident severity spectrum is divided into six categories (Wilmot et al., 1981), which are discussed further in Appendix D. The accident scenarios considered in this analysis fall into the three most severe of the six severity categories.

Accidents in the first three, least severe, categories result in no release of material from the spent nuclear fuel transportation cask. These categories include all the accident scenarios associated with handling the spent nuclear fuel cask, including dropping the cask during transfer from the vessel to the truck or train. The transportation casks are certified to maintain their integrity when dropped from 15 m (50 ft) onto a perfectly unyielding surface. During the cask transfer, however, the crane may lift the cask higher than 15 m (50 ft). As the dock surface is softer than "perfectly unyielding," the soft surface of the dock would compensate for the greater drop height. Studies (DOE, 1994m) have shown that a cask can be dropped from much higher than the certification test height onto a yielding surface, without breaching.

The accidents analyzed in the three highest severity categories include collision of vessels, either in the approach to the harbor or when the vessel transporting the foreign research reactor spent nuclear fuel would be docked. The category 4 accident severity category models a collision of two vessels resulting in the breach of the transportation cask. Severity categories 5 and 6 model collisions that would breach the cask and subsequent fires that would cause the release of additional material, with category 6 fires being more intense than those for category 5.

As mentioned above, the spectrum of accidents, including a container breach and fire, were evaluated at two locations in each of the 13 ports of entry selected to envelop the port impacts. The approach to each port, from the sea buoy to the selected dock, was examined to determine the location where the accident would be most likely as well as have the largest consequence. This point is typically near the highest population center along the approach to the pier, and DOE and the Department of State conservatively selected this point for accident analysis. The second location where the spectrum of accidents was assumed to occur is at pier-side.

At these two locations, the probability of an accident was assigned, based on historical ship accident data (see Appendix D for details). These data were used to develop accident frequencies for collisions between vessels large enough to generate the forces sufficient to damage the cask (additional details on the development of the model used are provided in Appendix D), and to develop the frequency of collisions concurrent with fires (Lloyds, 1991). These data include information on a large number of ship voyages and accidents due to all causes. The cause of the accident (human error, weather, mechanical failure, etc.) was not identified for this analysis. However, the data apply to damage to or loss of a vessel and would include information on accidents that were caused by severe weather. Although severe weather accident scenarios are not specifically identified in the analysis, they are considered through the use of these data.

The consequence modeling for the port accident analysis was performed using the MELCOR Accident Consequences Code System (MACCS) (Jow, 1990), a code developed for the conservative modeling of accident consequences for nuclear powerplants and approved by the NRC. This code uses site-specific information, including population and meteorology, along with the identified radionuclide inventory and release fractions to determine the consequences of the accident scenarios. In determining the effects of the release of radioactive material, the MACCS code evaluates the direct dose to the public as well as several additional pathways including inhalation, ingestion, and groundshine. Groundshine is the dose received from radioactive material deposited on the ground's surface.

A conservative assumption incorporated into the risk assessment is that the entire population would remain in the area for 24 hours and therefore would be exposed to the greatest extent possible to radioactive material deposited on the ground from the plume. In reality, individuals close to an accident could be evacuated.

Atmospheric dispersion is usually the primary mechanism for dispersing any material that might be released in a severe accident. For the ship-collision-without-fire scenario (category 4), the release is modeled as occurring at the water surface level. For shipboard fires (categories 5 and 6), an elevated release due to the lifting effect of the fire is modeled. Meteorology data from the nearest National Weather Service Station were obtained for the 13 ports of entry selected as representative ports and input to the analysis of dispersion to ensure validity.

Cask Characteristics

The behavior of the cask in accidents within each accident severity category is accounted for in this analysis. "Type B" spent nuclear fuel casks (the kind in which the foreign research reactor spent nuclear fuel would be shipped) are massive, highly damage-resistant packages. Moreover, the spent nuclear fuel itself consists mostly of solid metallic materials that are not readily dispersed. Therefore, large releases would not be likely to occur, even in the most severe of accident conditions.

"Type B" packages are required to pass a series of rigorous tests that are associated with hypothetical accident conditions that might be encountered. These certification tests were developed by the International Atomic Energy Agency and promulgated as model regulations (IAEA, 1990). These model regulations have been adopted by the United States as well as all of the nations currently proposing to ship foreign research reactor spent nuclear fuel to the United States under the basic implementation of Management Alternative 1.

Ports Selected for Accident Analyses

Analyses of the impacts of possible accidents at representative ports were conducted. Thirteen ports were selected as being representative of the full range of ports in the United States, based on population and geography. Three of the ports are high-population density ports, two on the East Coast (Elizabeth, NJ and Philadelphia, PA) and one on the West Coast (Long Beach, CA). Five of the ports (Portland, OR; Tacoma, WA; Concord NWS, CA; Jacksonville, FL; and Norfolk, VA) are medium-population density ports, three on the West Coast and two on the East Coast. The remaining ports (MOTSU, NC; Galveston, TX; Savannah, GA; Wilmington, NC; and Charleston, SC) are low-population density ports. The 13 potential ports of entry for which accidents were analyzed collectively have a range of populations and geography that ensure that the results of these analyses are representative of the results that would have been reached if the analyses had been conducted for all ports. Additionally, these 13 ports include all 10 of the ports that meet all of the port selection criteria.

To demonstrate the representative nature of the analyses performed, a plot was made of the analyzed accident consequences for mean meteorological conditions at each port versus the port's population in a 16-km (10-mi) radius (Figure 4-1). The analyzed data points are shown as dots. The straight line represents the linear least squares fit of the data. Since the straight line represents an average of the data, some deviation from the line for individual data points is expected. The data fit well, with a correlation factor of 0.994455 (a correlation factor of 1.0 implies a perfect fit). This plot demonstrates the expected increase in the total population dose with an increase in port population. Deviations from the line by the calculated data are typically due to the distribution of population in relation to the local meteorology. Where most of the population is downwind of the port in normal weather, the corresponding population dose would likely be above the average line. For comparison, the total population dose due to background radiation is shown in the upper right corner. This comparison shows that population dose resulting from a severe accident would be approximately 0.2 percent of the annual background population dose.

As a check that the data from the 16-km (10-mi) radius population is valid, a similar analysis was performed correlating the 80-km (50-mi) radius population and accident consequences for seven ports. This analysis confirmed that the population dose as a function of population is linear, and therefore confirms that the range of ports selected for accident analysis fully covers the entire range of U.S. ports. More specific discussion of the results of the analyses is provided in Appendix D. This linearity of consequences and population show that any port selected for use as an intermediate port or port of entry for the foreign research reactor spent nuclear fuel, ranging from the least populous port (MOTSU) to the most populous port (Elizabeth) and including major ports of intermediate population, has had representative accident analyses performed.

Probabilities of Port Accidents

The probability of an accident occurring can be determined from historical statistics on ship collisions and mishaps. Maritime accident rate data from a Lloyd's of London database covering approximately 900,000 port calls by commercial vessels over a 15-year period (1978 to 1993) were examined to develop accident probabilities. The data indicate that the basic accident rate in and near ports is slightly less than 0.0001 accidents per port transit, or approximately 1 accident per 10,000 port visits.

Only the most severe accidents, however, would cause any damage to the cask. Thus, the conditional probabilities of occurrence of each accident severity were also developed from this database. As discussed in Appendix D, a conditional probability is defined as the probability, given that an accident has occurred, that it will be of a certain severity. To calculate overall probability of an accident of a particular severity, the base accident probability (accident rate) must be multiplied by the conditional probability.

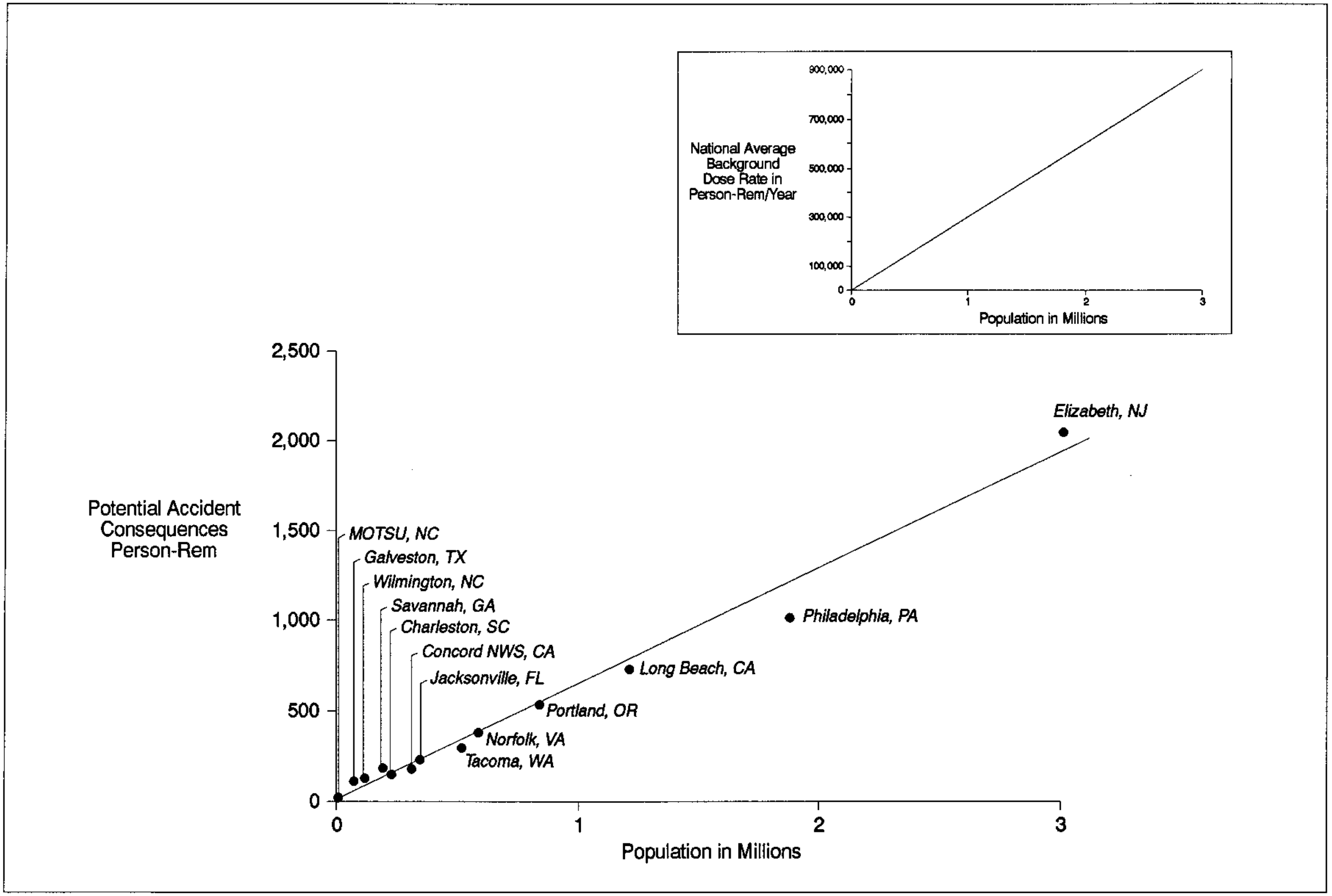


Figure 4-1 Consequences Versus Population [for a 16-km (10-mi) Radius]

Accidents are ranked according to their release categories. Release category 4 would result from the cask being damaged and compromised. Release category 5 would result from a damaged and compromised cask being enveloped in a fire. Release category 6 would result from a damaged and compromised cask being enveloped in a longer fire than a category 5 fire. The probabilities for the category 4, 5, and 6 accidents are 0.000006, 5×10^{-9}, and 6×10^{-10}, respectively. DOE and the Department of State assumed that it was equally likely that the accident occurs at the dock or in the channel, during the approach to the dock.

Consequences of Port Accidents

The consequence of an accident indicates the result, given that the accident were to occur, without any consideration for the likelihood of the accident occurring. The analysis conducted to determine the impacts of an accident involving foreign research reactor spent nuclear fuel in ports yields two different measures of the consequences. One measure is a calculation of the number of LCF that might result if the accident were to occur. These results are presented in Table 4-6 for the three most severe types of accidents under mean meteorological conditions.

The results presented in Table 4-6 are based on the mean consequences, so they are equivalent to results expected for the accidents in the respective release categories. These results are also based on the conservative assumption that accidents involve a cask carrying the highest inventory of nuclear material expected. Appendix D provides information on the consequences associated with the range of spent nuclear fuel types considered.

Examination of Table 4-6 shows that the most adverse consequence (2.9 LCF) arises from a Release category 5 accident in the channel approaching the Port of Elizabeth. This places the vessel just west (and generally upwind) of New York City. Although some of the release fractions change between categories 5 and 6, most of them do not. Therefore, the total population dose and the related number of LCF are about the same for Release categories 5 and 6. Release category 4 would be a release with no fire. In the absence of a fire, the release would remain at ground or water level without wide dispersion, hence the greatly reduced number of affected individuals and reduced consequences.

In addition to calculating the health effects of an accident on man, the MACCS code also calculates the impact of the accident on the land and structures around the accident site. These effects are characterized by the costs of activities required to bring the land and structures back into a usable condition. These activities are characterized as (1) no remedial action required; (2) decontamination – the resources can be returned to use immediately after clean-up; (3) interdiction – the resources must be temporarily abandoned, for several years, prior to their return to use; and 4) condemnation – the resources are considered unusable for an extended period. In all of the consequence analyses performed for each of the accident sites, there are no costs calculated that are associated with decontamination, interdiction, or condemnation. This means that all of the land and structures would be immediately available for use. (The consequences calculated by MACCS for the immediate vicinity of the accident are based on average value for the area within 1.6 km (1 mi) of the accident. Even though the average consequences calculated by MACCS show no costs associated with accident clean-up, the area immediately around the ship carrying the foreign research reactor spent nuclear fuel (i.e., the dock area) may require some remedial activity).

A sensitivity analysis was performed to address the potential impact of shipboard fires with extremely high temperatures that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum-based fuel or the combustion temperature of the TRIGA fuel. This analysis shows that the maximum consequences of such a fire are a factor of 100 larger than those

Table 4-6 Port Accident Consequences (LCF)

Locations	Accident Severity Category[a] 4	5	6
Elizabeth at the Dock	0.00010	2.8	2.7
Elizabeth in the Channel	0.00016	2.9	2.8
Long Beach at the Dock	0.000093	2.0	2.0
Long Beach in the Channel	0.000035	1.8	1.9
Philadelphia at the Dock	0.000078	1.2	1.2
Philadelphia in the Channel	0.000037	1.2	1.2
Portland at the Dock	0.000034	0.52	0.53
Portland in the Channel	0.000023	0.50	0.51
Norfolk at the Dock	0.000024	0.38	0.37
Norfolk in the Channel	0.000013	0.30	0.30
Charleston at the Dock (Wando Terminal)	0.000011	0.19	0.19
Charleston at the Dock (NWS Charleston)	0.0000068	0.22	0.22
Charleston in the Channel	0.000017	0.19	0.19
Tacoma at the Dock	0.000024	0.75	0.80
Tacoma in the Channel	0.000017	0.63	0.66
Concord NWS at the Dock	0.000019	0.90	0.96
Concord NWS in the Channel	0.000041	1.40	1.50
Jacksonville at the Dock	0.000012	0.31	0.31
Jacksonville in the Channel	0.000011	0.24	0.25
Savannah at the Dock	0.000025	0.23	0.23
Savannah in the Channel	0.0000059	0.18	0.19
Wilmington at the Dock	0.000017	0.22	0.23
Wilmington in the Channel	0.0000042	0.098	0.10
Galveston at the Dock	0.000032	0.64	0.70
Galveston in the Channel	0.000014	0.63	0.69
MOTSU at the Dock	0.0000032	0.099	0.11
MOTSU in the Channel	0.0000042	0.098	0.10

[a] *These accident release categories are the three highest in severity.*

calculated for the base case (Appendix D, Section D.5.4.2.2, Table D-31). An extremely high temperature ship fire is highly unlikely (one in ten billion per shipment) and the risks are comparable to those calculated in the base case. This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 m (1,000 ft). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

Risks

The calculated risk (probability times consequence) to the nearby population on a per-shipment basis assuming one cask per shipment and for the entire basic implementation of Management Alternative 1 is presented in Table 4-7. Each risk value is the sum of the risks from accident severity categories 4, 5, and 6. (A sensitivity study was performed to assess the risks associated with accidents that result in extremely high temperature fires. This sensitivity study was limited to an analysis of the per-shipment risks associated with shipment of spent nuclear fuel through the highest population density port, Elizabeth, NJ. Even though the consequences of this type of an accident are orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case. A more detailed comparison of the base case and sensitivity analyses is presented in Appendix D, Section 5.4.3.2.)

Table 4-7 Port Accident Risks

Port	*Per Shipment of One Cask*		*Total All Shipments*	
	Population Dose (person-rem)	*Population Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Elizabeth via:				
• Two High Population Ports	0.00013	5.6×10^{-8}	0.070	0.000029
• One High and One Intermediate Population Port	0.00011	4.8×10^{-8}	0.060	0.000025
• One High and One Low Population Port	0.00011	4.5×10^{-8}	0.057	0.000024
• Two Intermediate Population Ports	0.000056	2.4×10^{-8}	0.030	0.000013
• One Intermediate and One Low Population Port	0.000051	2.2×10^{-8}	0.027	0.000011
• Two Low Population Ports	0.000046	2.0×10^{-8}	0.024	0.000010
• Direct	0.000042	1.8×10^{-8}	0.022	0.0000094
Long Beach via:				
• Two High Population Ports	0.00011	4.7×10^{-8}	0.058	0.000025
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.038	0.000016
• Two Intermediate Population Ports	0.000050	2.1×10^{-8}	0.026	0.000011
• One Intermediate and One Low Population Port	0.000041	1.8×10^{-8}	0.022	0.0000092
• Two Low Population Ports	0.000032	1.4×10^{-8}	0.017	0.0000072
• Direct	0.000028	1.2×10^{-8}	0.015	0.0000062
Philadelphia via:				
• Two High Population Ports	0.00011	4.5×10^{-8}	0.057	0.000024
• One High and One Intermediate Population Port	0.000088	3.7×10^{-8}	0.047	0.000020
• One High and One Low Population Port	0.000083	3.5×10^{-8}	0.044	0.000019
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.016	0.0000072
• One Intermediate and One Low Population Port	0.000026	1.1×10^{-8}	0.014	0.0000061
• Two Low Population Ports	0.000021	9.3×10^{-9}	0.011	0.0000049
• Direct	0.000017	7.5×10^{-9}	0.0092	0.0000040
Portland via:				
• Two High Population Ports	0.000090	3.8×10^{-8}	0.047	0.000020
• One High and One Intermediate Population Port	0.000059	2.5×10^{-8}	0.031	0.000013
• One High and One Low Population Port	0.000050	2.2×10^{-8}	0.027	0.000011
• Two Intermediate Population Ports	0.000029	1.3×10^{-8}	0.015	0.0000066
• One Intermediate and One Low Population Port	0.000020	9.0×10^{-9}	0.011	0.0000047
• Two Low Population Ports	0.000011	5.1×10^{-9}	0.0059	0.0000026
• Direct	0.0000073	3.2×10^{-9}	0.0039	0.0000017
Norfolk via:				
• Two High Population Ports	0.000095	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000076	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000071	3.0×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000019	8.3×10^{-9}	0.0098	0.0000044
• One Intermediate and One Low Population Port	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Two Low Population Ports	0.0000088	4.0×10^{-9}	0.0046	0.0000021
• Direct	0.0000048	2.1×10^{-9}	0.0025	0.0000011
Charleston (Wando Terminal) via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.4×10^{-9}	0.0087	0.0000039

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
• One Intermediate and One Low Population Port	0.000012	5.2×10^{-9}	0.0061	0.0000027
• Two Low Population Ports	0.0000066	3.1×10^{-9}	0.0035	0.0000016
• Direct	0.0000027	1.2×10^{-9}	0.0014	6.4×10^{-7}
Charleston (NWS Charleston) via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000017
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0084	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0058	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0032	0.0000016
• Direct	0.0000028	1.3×10^{-9}	0.0011	6.8×10^{-7}
Galveston via:				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000080	3.4×10^{-8}	0.042	0.000018
• One High and One Low Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• Two Intermediate Population Ports	0.000023	1.0×10^{-8}	0.012	0.0000053
• One Intermediate and One Low Population Port	0.000018	8.0×10^{-9}	0.0094	0.0000042
• Two Low Population Ports	0.000013	5.8×10^{-9}	0.0068	0.0000031
• Direct	0.0000090	4.0×10^{-9}	0.0047	0.0000021
Jacksonville via:				
• Two High Population Ports	0.000094	4.0×10^{-8}	0.050	0.000021
• One High and One Intermediate Population Port	0.000075	3.2×10^{-8}	0.040	0.000017
• One High and One Low Population Port	0.000070	2.9×10^{-8}	0.037	0.000016
• Two Intermediate Population Ports	0.000018	7.9×10^{-9}	0.0093	0.0000041
• One Intermediate and One Low Population Port	0.000013	5.7×10^{-9}	0.0067	0.0000030
• Two Low Population Ports	0.0000078	3.6×10^{-9}	0.0041	0.0000019
• Direct	0.0000038	1.7×10^{-9}	0.0020	9.0×10^{-7}
Savannah via:				
• Two High Population Ports	0.000093	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000074	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000017	7.5×10^{-9}	0.0088	0.0000039
• One Intermediate and One Low Population Port	0.000012	5.3×10^{-9}	0.0062	0.0000028
• Two Low Population Ports	0.0000068	3.2×10^{-9}	0.0036	0.0000017
• Direct	0.0000028	1.3×10^{-9}	0.0015	6.9×10^{-7}
Wilmington via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000073	3.1×10^{-8}	0.039	0.000016
• One High and One Low Population Port	0.000068	2.9×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000016	7.2×10^{-9}	0.0084	0.0000038
• One Intermediate and One Low Population Port	0.000011	5.0×10^{-9}	0.0058	0.0000026
• Two Low Population Ports	0.0000062	2.9×10^{-9}	0.0032	0.0000015
• Direct	0.0000022	1.0×10^{-9}	0.0012	5.3×10^{-7}
Tacoma via:				
• Two High Population Ports	0.000092	3.9×10^{-8}	0.049	0.000021
• One High and One Intermediate Population Port	0.000062	2.6×10^{-8}	0.033	0.000014
• One High and One Low Population Port	0.000053	2.3×10^{-8}	0.028	0.000012
• Two Intermediate Population Ports	0.000031	1.4×10^{-8}	0.017	0.0000072
• One Intermediate and One Low Population Port	0.000023	1.0×10^{-8}	0.012	0.0000053
• Two Low Population Ports	0.000014	6.1×10^{-9}	0.0072	0.0000032
• Direct	0.0000097	4.3×10^{-9}	0.0051	0.0000023

Port	Per Shipment of One Cask		Total All Shipments	
	Population Dose (person-rem)	Population Risk (LCF)	Population Dose (person-rem)	Population Risk (LCF)
Concord NWS via				
• Two High Population Ports	0.000099	4.2×10^{-8}	0.052	0.000022
• One High and One Intermediate Population Port	0.000069	2.9×10^{-8}	0.036	0.000015
• One High and One Low Population Port	0.000060	2.5×10^{-8}	0.032	0.000013
• Two Intermediate Population Ports	0.000038	1.7×10^{-8}	0.020	0.0000087
• One Intermediate and One Low Population Port	0.000029	1.3×10^{-8}	0.016	0.0000067
• Two Low Population Ports	0.000021	9.0×10^{-9}	0.011	0.0000047
• Direct	0.000017	7.1×10^{-9}	0.0088	0.0000038
MOTSU via:				
• Two High Population Ports	0.000091	3.9×10^{-8}	0.048	0.000020
• One High and One Intermediate Population Port	0.000072	3.1×10^{-8}	0.038	0.000016
• One High and One Low Population Port	0.000067	2.8×10^{-8}	0.036	0.000015
• Two Intermediate Population Ports	0.000015	6.8×10^{-9}	0.0080	0.0000036
• One Intermediate and One Low Population Port	0.000010	4.6×10^{-9}	0.0054	0.0000024
• Two Low Population Ports	0.0000053	2.5×10^{-9}	0.0028	0.0000013
• Direct	0.0000013	6.2×10^{-10}	0.00069	3.2×10^{-7}

The column on the left indicates the port of entry for the foreign research reactor spent nuclear fuel and the possible combinations of two intermediate ports, as well as no intermediate ports (direct). The second and third columns present two measures of the risk on a per-shipment basis. These risks are based on the conservative assumption that all East and West Coast deliveries would follow the same route for all shipments. The fourth and fifth columns sum the per-shipment risks for all of the shipments, for the entire basic implementation of Management Alternative 1. The two columns under the heading of "Total All Shipments" are the product of the per-shipment risk data for each type of spent nuclear fuel cask with the number of casks of that spent nuclear fuel type. DOE and the Department of State conservatively assumed in these calculations that each port would receive all the casks.

Consider first the per-shipment population exposure risk for a shipment of foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports. This value, 0.00013 person-rem, is the risk from one cask shipment of the highest nuclear material inventory which would first pass through two high-population density ports, such as Boston and Philadelphia, then would be delivered to Elizabeth. The risk of this cask shipment would be the sum of the risks associated with each of the three ports, because an accident could occur in any of the ports. Since risk is the product of consequences and probability, and probability has no units, the risk would be expressed in the units of the consequences, in this case population exposure (person-rem).

Comparing the risk of sending the foreign research reactor spent nuclear fuel to Elizabeth via two high-population density ports (0.00013 person-rem) to sending the spent nuclear fuel directly to Elizabeth (0.000042 person-rem) shows that the risk would be cut by about two-thirds by eliminating the intermediate ports. This is expected, since the estimated overall risk is the sum of the risk at each of the three ports, and three high-population density ports would have roughly the same risks. Now compare the per-shipment risk of using a low-population density port, say MOTSU, via two low-population density ports. Table 4-7 indicates that this risk would be 0.0000053 person-rem, or about 25 times lower than the highest risk, Elizabeth via two high population ports. All per-shipment risks are conservatively based on the highest nuclear material inventory cask to maximize the potential risk.

The manner of evaluating the per-shipment risk of LCF in Table 4-7 is the same as for the per-shipment population exposure risk. Once again, shipping the foreign research reactor spent nuclear fuel through or into high-population density ports would increase the risk, as would using ships that pass through intermediate ports on their way to the port of entry.

The range of total population risks would be from 0.070 to 0.00069 person-rem for the population dose and from 0.000029 to 3.2×10^{-7} LCF for the risk, comparing shipping to Elizabeth via two high-population density ports and shipping to MOTSU without intermediate ports. The highest estimated population risk due to port accidents that might occur due to the basic implementation of Management Alternative 1 is 0.000029 LCF. This means that there would be less than a one in ten thousand chance of some member of the public incurring an LCF due to the basic implementation of Management Alternative 1 port transits.

The highest estimated MEI accident risk is conservatively determined by multiplying the accident probability by the consequences, in terms of dose to the MEI, of that accident. The MEI in this case is assumed to be an individual at the center of the plume less than 1.6 km (1 mi) from the accident. The highest average MEI doses calculated for the accident severity categories are: 0.11 mrem for category 4, 117 mrem for category 5, and 95 mrem for category 6. See Appendix D, Section D.5.4.2.2 for details. The reason MEI dose for category 6 is relatively lower than that for category 5 is because the larger category 6 associated fire would disperse the radioactive material faster and farther than the category 5 fire. For the 721 shipments in the basic implementation of Management Alternative 1, and using the per port transit accident probabilities in Appendix D, the highest MEI accident risk is estimated to be 0.00042 mrem. This corresponds to about 2×10^{-10} LCF. This means that the chance of the MEI incurring an LCF due to a port accident under the basic implementation of Management Alternative 1 would be less than one in a billion.

Emergency Management and Response

Emergency response capabilities for a foreign research reactor spent nuclear fuel mishap would be available through the U.S. Coast Guard and the local jurisdictions surrounding each candidate port of entry, with specialized support available from DOE. The specialized analysis and identification of potential hazards, use of the robust "Type B" packaging, specific emergency plan and procedure development, training, response rehearsal, and interagency coordination for efficient and effective response would minimize the potential consequences should a foreign research reactor spent nuclear fuel mishap occur. The specific emergency management and response capabilities and responsibilities are described in Chapter 2, Section 2.7.

At military ports, the U.S. Coast Guard routinely provides safety/security screen escorts. The addition of foreign research reactor spent nuclear fuel shipments would have almost no effect on their ongoing operations.

Consequences of Port Accidents

A sensitivity analysis was performed to address the potential impact of extremely high temperature fires, fires that could result in the foreign research reactor spent nuclear fuel attaining temperatures above the melting point of the aluminum based fuel or the combustion temperature of the TRIGA fuel, on the consequences of an accident in port. This analysis, which uses the Port of Elizabeth, NJ as the site of the accident, is presented in Appendix D, Section D.5.4.3.2, and shows that even though the consequences of this type of an accident are two orders of magnitude larger than those calculated for the base case analysis, this type of event is highly improbable and the risks are comparable to those calculated in the base case.

This analysis also addressed the impact of an accident on the land around the port. Using the same characterizations as described in the preceding paragraph, the largest mean impact distance is a decontamination and interdiction distance which is limited to approximately 300 meters (1000 feet). The analysis also shows that this distance is representative of the impact of this highly improbable type of accident at any of the ports included in the proposed action.

4.2.2.4 Cumulative Impacts of Port Activities

Port workers are expected to be exposed to other shipments of radioactive materials in addition to those associated with the basic implementation of Management Alternative 1. These shipments include DOE and commercially initiated programs. An assessment has been made of the cumulative impact of the incident-free dose to the maximally exposed worker from all of these activities. The cumulative analysis is based on data collected at several ports for 2.5 years (January 1992 through June 1994). The maximally exposed port worker is estimated to receive less than 10 mrem per year from commercial shipments. Details of this analysis are presented in Appendix D, Section D.4.6. As previously stated, based on cask dose rates equal to the regulatory limit, the maximally exposed port worker could receive an annual dose greater than the NRC and DOE regulatory limit of 100 mrem per year (NRC, 1991). Therefore, DOE would implement mitigation measures.

4.2.2.5 Port Activities Mitigation Measures

As with marine transport, the principal environmental impact that would occur during port activities is radiation dose to workers. No members of the general public would be close enough to the transportation cask to receive any radiation dose. The workers would receive this dose during safety inspections and handling activities which cannot be curtailed.

Two conservative assumptions in this analysis drive the maximally exposed worker dose higher than would actually be expected. The radiation dose rate near every foreign research reactor spent nuclear fuel shipping container is assumed to be equal to the regulatory limit and the same individual is assumed to conduct all the inspections. Neither of these is actually likely to occur.

Nevertheless, DOE and the Department of State would require, through a clause in the shipping contracts, some administrative controls on the port workers to mitigate the radiation doses to the workers during inspection and handling activities. DOE and the Department of State would implement a system to track the inspectors and other port workers actually involved in the shipment of foreign research reactor spent nuclear fuel. If any inspector's or worker's dose approaches 100 mrem in any year, then DOE and the Department of State would require other inspectors or workers to be used. In this way, the maximally exposed worker dose would be constrained to the regulatory limit.

If a cask or casks were sunk in coastal waters, DOE and the Department of State would employ modern underwater search techniques to locate and recover the cask(s), thus minimizing the potential impacts to marine life.

4.2.2.6 Environmental Justice at the Port(s)

Executive Order 12898 deals with the issue of environmental justice and directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations.

The concept of environmental justice is discussed in more detail in Appendix A. During normal port activities associated with receipt of the foreign research reactor spent nuclear fuel shipments—including harbor activities, unloading the ship, transfer of the spent nuclear fuel containers to truck or train, and movement out of the port city—the dominant radiological impacts have been shown to be the exposures received by the workers in the immediate vicinity of the shipping container. These individuals include the inspectors, shipping container handlers, truck drivers, etc. Since the intensity of the gamma radiation falls off rapidly with distance, the doses that might be received by other workers and members of the general population can in theory be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks would be ruptured and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few kilometers downwind of the fire but small amounts could be carried out for several tens of kilometers.

Because the particular details of both the accident conditions (such as the severity of a fire) and the weather conditions at the time of an accident could vary so much, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accidents (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). The risk of LCF was found to be so small that zero LCF would be expected due to accidents at ports.

Appendix A describes minority populations and low-income households residing near the ports. Calculations for incident-free and accident conditions clearly demonstrate that for the general population, including minority and low-income groups, the radiological impacts would be very low. Minority or low-income populations living near the potential ports of entry would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to the same very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at candidate ports, including the social and economic status of the general population, minority populations, and the low-income population surrounding candidate ports. Economic benefits that would result from increased cargo handling and transportation in the port area would be extremely small for the general population or any particular segment of the population residing near candidate ports.

4.2.3 Ground Transport Impacts

Foreign research reactor spent nuclear fuel is transported in large, heavy containers called transportation casks. Transportation casks are designed and constructed to contain the radioactivity in spent nuclear fuel during severe transportation accidents. NRC has estimated that transportation casks will withstand 99.4 percent of truck and rail accidents without sustaining damage sufficient to breach the transportation cask (NRC, 1987). Only in the worst conceivable conditions, which are of low probability, could a transportation cask of the type used to transport spent nuclear fuel be so damaged that there is a reasonable possibility of release of radioactivity to the environment.

Spent nuclear fuel has been transported along highways, railways, and waterways since 1949. Federal standards describe the routing requirements for spent nuclear fuel shipments. Spent nuclear fuel transported includes foreign research reactor, commercial, naval, and DOE spent fuel. Since 1949, there have been 21 incidents involving vehicles carrying irradiated fuel elements. None of these incidents resulted in damage to the structural integrity of a cask or the release of the cask's contents.

4.2.3.1 Conservative Assumptions and Analytic Approach

Transportation impacts may be divided into two parts: the impacts due to incident-free transportation and the impacts due to transportation accidents. For incident-free transportation and transportation accidents, impacts may be further divided into two parts: nonradiological impacts and radiological impacts. The nonradiological impacts consist of the vehicular impacts of transportation, such as vehicular emissions and traffic accidents.

For incident-free transportation, the radiological impacts would result from the radiation field that surrounds the cask. For transportation accidents, the radiological impacts would be based on the radioactivity released from the spent nuclear fuel transportation cask during the accident. Impacts are estimated for workers and the population along the transportation route.

For both incident-free transportation and transportation accidents, methodology developed by NRC and used by DOE in the *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement* (SNF&INEL Final EIS) (DOE, 1995c) was used to estimate the impacts for foreign research reactor spent nuclear fuel in this EIS. These impacts were quantified as the estimated number of radiation-related cancer fatalities and the estimated number of nonradiological fatalities from vehicular emissions and traffic accidents. Appendices B, C, D, E, and F of this EIS contain more details on the equipment, regulations, and experience associated with spent nuclear fuel transportation, and the methodology, data, and conservative assumptions used to develop these estimates.

Under the basic implementation of Management Alternative 1, acceptance of the foreign research reactor spent nuclear fuel would require the transport of approximately 837 casks from seaports and Canadian border crossings to DOE facilities. The number of casks was determined by assigning the spent nuclear fuel from each foreign research reactor to a reasonably available and capable cask. Conservative assumptions were used to estimate cask capacity, which is based on physical, thermal, and radiological characteristics of the spent nuclear fuel. Appendix B contains more details on the foreign research reactor spent nuclear fuel transportation casks.

For the purposes of analysis in this EIS, the initial ground transportation activities to the Savannah River Site and/or the Idaho National Engineering Laboratory is called Phase 1, and the possible subsequent intersite ground transport and continued management is called Phase 2. The impact assessment includes analysis of between 13 and 161 intersite shipments, depending upon the mode of transportation (truck or rail) and the potential foreign research reactor spent nuclear fuel management sites that might be selected. Intersite shipments would be fewer than foreign shipments because the spent nuclear fuel would be cooler. Larger casks would likely be used, and more foreign research reactor spent nuclear fuel would be consolidated.

The first step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis assuming one cask per shipment, for transportation of the various spent nuclear fuel casks. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities and the magnitudes of exposure are discussed in Appendix E. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask, and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure and the magnitudes of exposure are discussed in Appendix E.

Calculation of risk factors was accomplished by first using the HIGHWAY (Johnson, et al., 1993a) and INTERLINE (Johnson, et al., 1993b) computer codes to choose representative routes in accordance with the U.S. Department of Transportation regulations. These codes provide population estimates along the routes so that the RADTRAN (Neuhauser, 1993) and RISKIND (Yuan, et al., 1993) codes could be used to determine the risk factors associated with ground transportation activities. These computer codes are described in more detail in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and Appendix E of this EIS.

The single largest contributor to the ground transport population doses (about 80 percent) calculated with RADTRAN was found to be the dose to members of the public at truck stops. The parameters used to calculate doses during truck stops are quite conservative. The parameters are based on the assumption that stops occur as a function of distance, with a truck stop rate of 0.011 hr per km (0.018 hr per mi). This stop rate results in over an hour of stop time per 100 km (62 mi) of travel. It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). These parameters were used because they are the default parameters in the RADTRAN code and they were used in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These conservative assumptions that are built in the code are highly unlikely to occur.

The next step is to use the risk factors and the number of shipments to estimate the risk of every possible way the foreign research reactor spent nuclear fuel program could be implemented. Because of the large number of ports, cask types, spent nuclear fuel types, and implementation options, simplifying assumptions are needed to control the amount of repetitive analysis:

- A review of the accident risk factors for the various types of spent nuclear fuel (see Appendix E) indicates that there is relatively little variation between the different types of foreign research reactor spent nuclear fuel, thus, it is not overly conservative to use the highest risk factors for all shipments.

- Spent nuclear fuel from countries bordering the Atlantic Ocean and Mediterranean Sea was assumed to arrive on the East Coast of the United States. Spent nuclear fuel from countries bordering the Indian and Pacific Oceans was assumed to arrive on the West Coast. This is conservative from an overland transportation standpoint, because, as shown in Appendix E, marine shipment to the coast nearest the management site would reduce the risk factors for the overland shipment.

- To account for the return transport of empty casks, the impacts due to vehicle emissions and traffic accidents were multiplied by two.

The foreign research reactor spent nuclear fuel could arrive at any of the ports of entry selected by DOE and the Department of State using criteria that are detailed in Appendix D, and would be likely to arrive at a variety of these ports. Therefore, the proposed impacts were completely analyzed three times, consisting of an upper bounding case, a lower bounding case, and an average case, for both truck and rail shipments. The upper bound case conservatively assumes the port(s) with the highest risk factors was chosen for each transportation activity. The risk factors are generally a function of distance and total population along the port to management site route, so the port chosen often shifted between Phase 1 and Phase 2. Conversely, the lower-bound case assumes ports with the lowest risk factors.

The average case is designed to provide a realistic estimate of the ground transport risk of transporting the foreign research reactor spent nuclear fuel. The risk factors are an arithmetic average of the risk factors for all acceptable ports. This represents the risk associated with the basic implementation of Management Alternative 1 and receiving foreign research reactor spent nuclear fuel at a variety of commercial ports.

Since each potential port of entry and each management site is capable of receiving spent nuclear fuel via rail or highway, the program was analyzed using each mode of transportation. The exception to this is the Nevada Test Site which has no existing rail capability, so that link was approximated by a hypothetical rail line to the Yucca Mountain Site. Additionally, the potential to use trucks to carry the relatively small casks from ports to potential foreign research reactor spent nuclear fuel management sites and rail to carry larger casks between potential foreign research reactor spent nuclear fuel management sites was analyzed. Site to site shipment would not occur until approximately 2006, so it is difficult to precisely predict which cask would be used. The analysis is based on a truck cask that carries 4 times as much spent nuclear fuel as a foreign cask, and a rail cask that carries 10 times as much spent nuclear fuel.

4.2.3.2 Impacts of Incident-Free Ground Transport

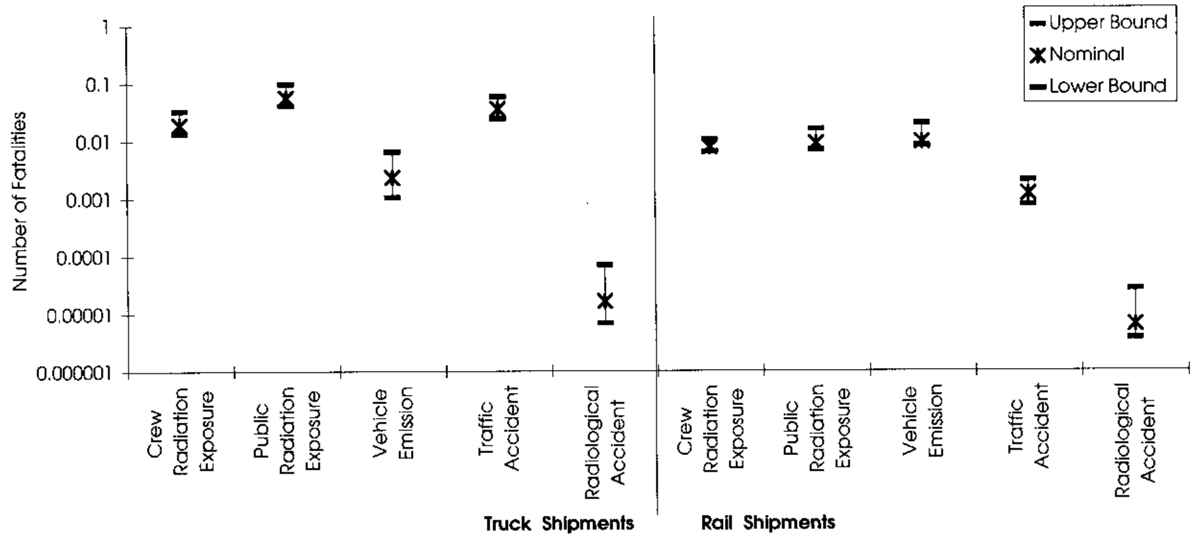
The incident-free transportation of spent nuclear fuel was estimated to result in total population risk that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the transportation workers. Thus, the calculated maximum risk value for overland transportation is less than one fatality from cancer due to the basic implementation of Management Alternative 1. The range of fatality estimates is caused by two factors: (1) the option of using truck or rail to transport spent nuclear fuel; and (2) combinations of Phase 1 and Phase 2 sites that create varying cask shipment numbers and distances.

The estimated number of LCF due to radiation exposure for transportation workers ranged from 0.006 to 0.071. The estimated number of radiation-related LCF for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.052. These incident-free results apply to the workers and the public because both would be close enough to the cask to receive some radiation dose.

The impacts of transportation which are based on four Programmatic SNF&INEL Final EIS (DOE, 1995c) programmatic alternatives are summarized in Figures 4-2 through 4-5. The impacts of these additional programmatic alternatives are described in more detail in Appendix E.

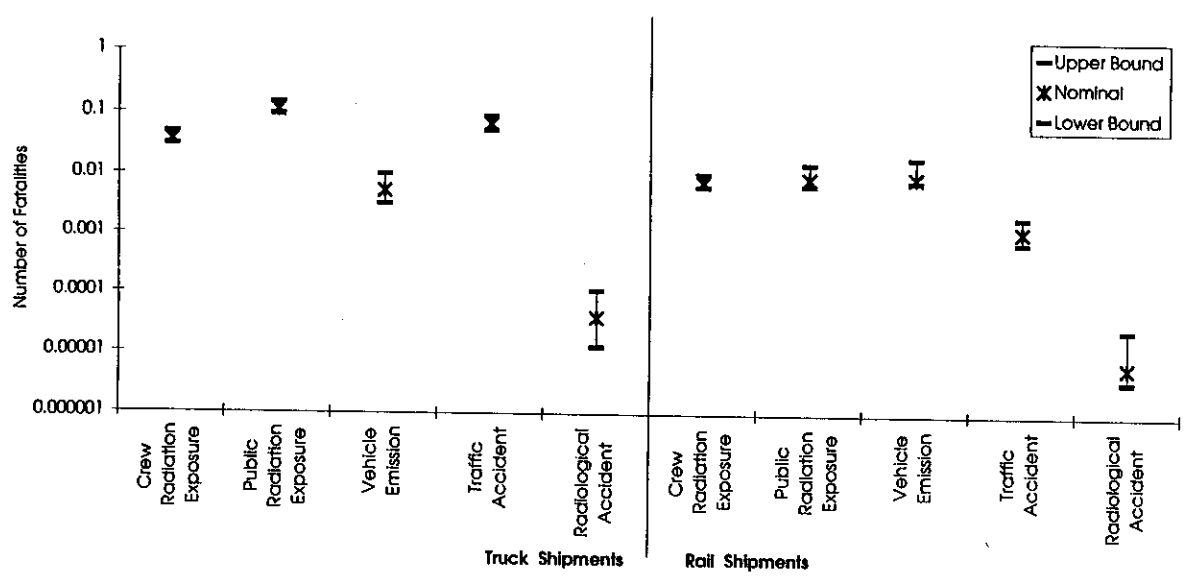
The highest estimated ground transport maximally exposed worker risk is 0.00052 LCF, just like the marine transport and port worker risks. This estimate is based on the conservative assumption that one truck driver makes enough trips to reach the regulatory limit of 100 mrem per year every year for 13 years. This means that under the assumptions described above, the chance of this individual incurring an LCF due to the basic implementation of Management Alternative 1 would be less than one in a thousand.

The highest estimated incident-free population risk is 0.30 LCF, which means that there would be a 30 percent chance of one additional cancer fatality among the public and the ground transport workers due to the basic implementation of Management Alternative 1.



- *Analysis of Decentralization applies to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-2 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-3 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

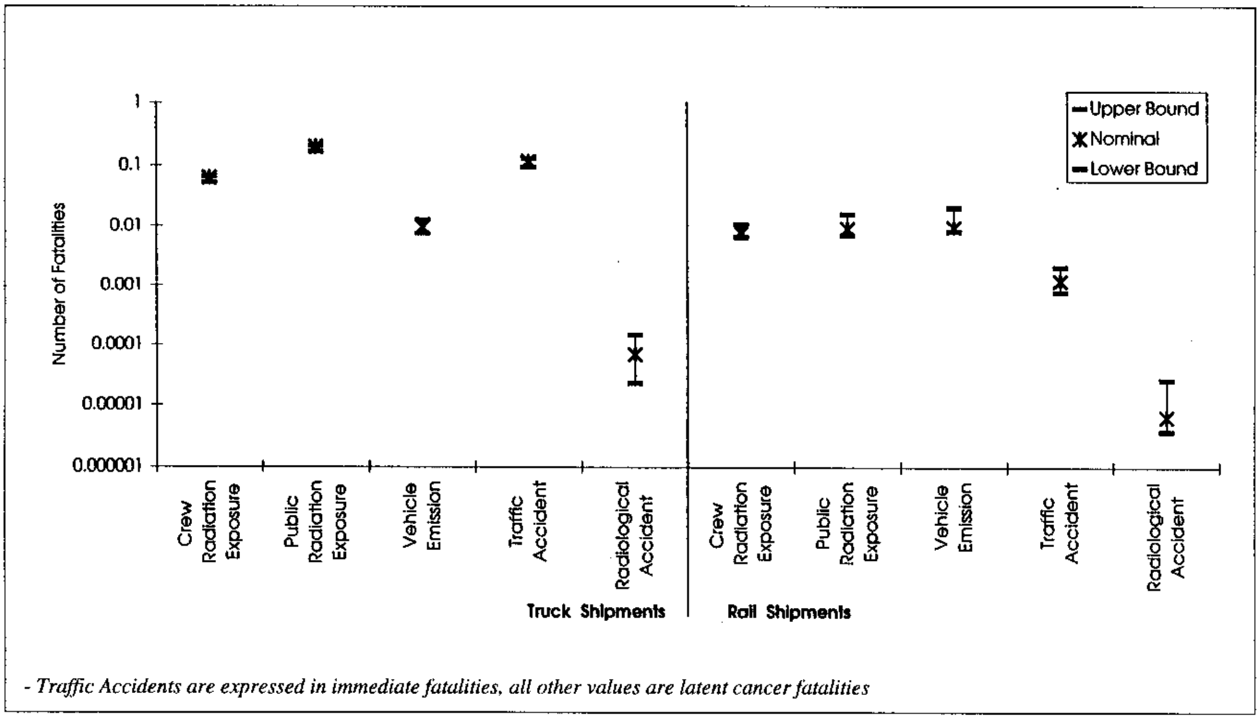


Figure 4-4 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

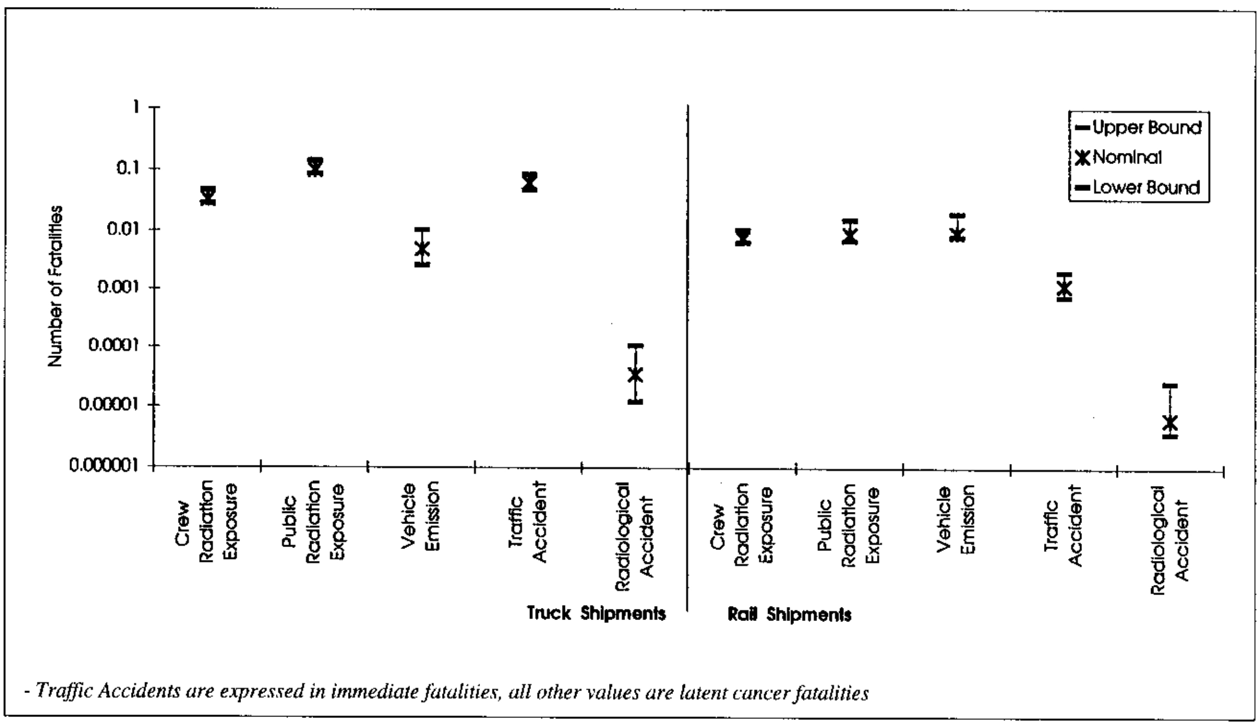


Figure 4-5 Range of Estimated Fatalities (Latent and Immediate) Under Basic Implementation of Management Alternative 1 and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

4.2.3.3 Impacts of Accidents During Ground Transport

The most severe accidents that might reasonably occur on this leg of the journey are truck or train crashes, followed by a large fire. If an accident occurred on a causeway at or near a port that caused a cask to fall into seawater, the consequences would be the same as if a cask fell off a ship into seawater. These consequences are presented in Section 4.2.1.3 under the subheading "Sunken Cask." Each State, and most local jurisdictions, maintain a hazardous materials response capability and a radiological protection program. These capabilities, along with the DOE radiological response assets that would be on-call for immediate technical assistance and response, would provide a high-level of expertise and would reduce the potential impacts of a foreign research reactor spent nuclear fuel accident.

Since hazardous materials team training is required to include radiological materials response, each team possesses a basic level of understanding and capability for a foreign research reactor spent nuclear fuel incident response. An incremental enhancement for spent nuclear fuel-specific response characteristics and planning may be required, especially for those jurisdictions along selected routes whose emergency responders are primarily volunteer organizations.

The development of a transportation plan specifically for the shipping campaign that would incorporate and integrate State and local emergency response plans, would increase emergency responder effectiveness and reduce the potential consequences of a foreign research reactor spent nuclear fuel accident.

Each State's emergency planning infrastructure, using the Local Emergency Planning Committees to the State Emergency Response Commission, enables these jurisdictions to identify and resolve potential emergency management and response issues and communicate issues that would require DOE and Department of State attention. This, along with DOE's Transportation External Coordination/Working Group, would ensure that all concerned agencies would be involved in the planning process to address potential problems before they become major hazards.

Risks

The total ground transportation accident risks for the basic implementation of Management Alternative 1 are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and potential foreign research reactor spent nuclear fuel management sites that might be selected. Section 4.10 compares these risks to those of common activities. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The risk of 0.14 for a traffic fatality means that under these conservative assumptions there would be a 14 percent chance of a traffic fatality related to the basic implementation of Management Alternative 1.

The maximum foreseeable offsite transportation accident would involve a shipment of foreign research reactor spent nuclear fuel in a suburban population zone under neutral (average) weather conditions. The accident has a probability of occurrence of about 0.0000001 per year (one chance in ten million), and could result in 14 person-rem and no fatalities. The probability of an accident occurring is at least an order of magnitude smaller in either an urban area or under stable atmospheric conditions. The consequences are less than an order of magnitude larger.

The impacts of transportation accidents are summarized in Figures 4-2 through 4-5, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk under each representative Programmatic SNF&INEL Final EIS alternative.

The highest estimated MEI radiological risk to members of the public due to accidents during ground transport is 1.4 x 10^{-11} LCF. This means that the chance of this individual incurring a cancer due to the basic implementation of Management Alternative 1 would be less than one in ten billion.

The highest estimated population radiological risk due to accidents is 0.00028 LCF, which is much less than one LCF.

4.2.3.4 Ground Transport Cumulative Impacts

The Programmatic SNF&INEL Final EIS (DOE, 1995c) analyzed the cumulative impacts of ground transportation, taking into account impacts from: (1) historical shipments of spent nuclear fuel to the five proposed foreign research reactor spent nuclear fuel management sites; (2) the programmatic alternatives; (3) other reasonably foreseeable actions that include transportation of radioactive material; and (4) general radioactive materials transportation that is not related to a particular action. The transportation of foreign research reactor spent nuclear fuel is included in the calculated totals under the spent nuclear fuel shipments for the Programmatic SNF&INEL Final EIS Alternatives 1 through 5. Proposed transportation of all spent nuclear fuel (of which the foreign research reactor fuel is a small component) accounts for less than one percent of the total LCF attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one quarter of that one percent. The total number of LCF over the time period 1943 through 2035 was estimated to be 290.

4.2.3.5 Ground Transport Mitigation Measures

The principal environmental impacts that would occur during ground transport are: (1) LCF due to radiation exposure, (2) LCF due to vehicular emissions, and (3) immediate fatalities due to traffic accidents. All three of these would be reduced by choosing port(s) of entry close to the management site(s). This would minimize the distance that must be covered by the vehicle(s).

Furthermore, in the case of truck transport, the truck driver(s) would be monitored for radiation dose. The annual maximally exposed worker limit of 100 mrem would never be approached during any single shipment, but the same driver could be used for multiple shipments throughout a year. DOE would implement mitigation measures through the foreign research reactor spent nuclear fuel acceptance contracts to ensure that each individual driver's dose remains below the regulatory limit. If any individual truck driver accumulates a dose approaching this limit in a year, DOE would require that new driver(s) be used to keep each individual driver's dose below the regulatory limit.

Since the casks would produce a radiation field of less than 10 mrem/hr at 2 m (6.6 ft) from the vehicle, an individual member of the general public would have to be within 2 m (6.6 ft) of the vehicle for at least ten hours in a year to receive a dose equal to the regulatory limit of 100 mrem/yr. A truck is not likely to sit in a traffic jam right beside another vehicle for as long as ten hours and an individual gas station attendant is not likely to spend ten hours refueling the trucks carrying foreign research reactor spent nuclear fuel. Therefore, DOE does not plan to implement ground transport mitigation measures for members of the general public.

4.2.3.6 Barge Transport

DOE and the Department of State have examined the possibility of using barges for the transport of foreign research reactor spent nuclear fuel as a substitute for truck or rail transport. The only two locations where barge transport is feasible are from the Port of Portland, OR up the Columbia River to the Hanford

Site and from the Port of Savannah, GA up the Savannah River to the Savannah River Site. Barge transport could only be implemented if one or both of these port/site combinations is selected in the Record of Decision.

For barge transport up the Columbia River, the incident-free radiological risk to the public would be approximately 0.0000043 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000029 and 0.0000058 LCF per shipment, respectively. For barge transport up the Savannah River, the incident-free radiological risk to the public would be approximately 0.0000019 LCF per cask shipment. This is slightly lower than the similar truck and rail shipment risks, which would be 0.000028 and 0.0000026 LCF per shipment, respectively.

For barge transport up the Columbia River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 3.5×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 1.5×10^{-8} and 3.8×10^{-9} LCF per shipment, respectively. For barge transport up the Savannah River, the accident radiological risk due to both airborne and waterborne pathways would be approximately 2.9×10^{-8} LCF per cask shipment. This is slightly higher than the similar truck and rail shipment risks, which would be 9.4×10^{-9} and 1.1×10^{-9} LCF per shipment, respectively.

The barge transport analysis is presented in more detail in Appendix E, Section E.8.15. The net result is that the foreign research reactor spent nuclear fuel could be transported by barge with approximately the same level of risk to workers and the public as if it was transported by truck or rail.

4.2.3.7 Environmental Justice Along Ground Transport Routes

The dominant radiological risks and impacts associated with incident-free transportation activities are the exposures received by the workers in the immediate vicinity of the casks and people who might be near the casks at truck stops. These individuals would be the only people receiving a measurable exposure during a spent nuclear fuel shipment. As discussed in Section 4.2.3.2, the number of radiation-related latent cancer deaths among transportation workers and the general public combined was calculated to be less than one. The same is true for cancer due to vehicle emissions. Ground transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential impacts from low probability accidents vary considerably and are dependent on the accident conditions (such as the size of the resulting fire, if any) and the weather conditions at the time of an accident. Transportation accidents were estimated to result in no LCF due to radiation and less than 0.2 immediate deaths due to traffic fatalities (see Section 4.2.3.3).

As described in Appendix A, the percentage of the total population comprised of minorities or low-income households varies among routes. Calculations for incident-free and accident conditions demonstrate that for the general population the radiological impacts would be very low. Minority or low-income populations living near these routes would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts as would the general population.

Implementation of the proposed action would have extremely low nonradiological effects on the environment along transportation routes, including the social and economic status of the general population, minority populations, and the low-income population residing along the transportation routes. Economic benefits that would result from increased transportation of cargo along transportation routes would be extremely small for the general population or any particular segment of the population residing along the transportation routes.

4.2.4 Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section presents the potential environmental impacts from the basic implementation of Management Alternative 1 at the potential foreign research reactor spent nuclear fuel management sites, namely the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site. It summarizes the detailed site analysis presented in Appendix F, Sections F.4, F.5, and F.6. The analysis examined environmental topics such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational and public health and safety, noise, traffic and transportation, utilities and energy, and waste management. The analysis showed that the basic implementation of Management Alternative 1 would not have a major effect on any of the environmental topics. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

Because of the public interest in radiation exposure to workers and the public, Section 4.2.4.1 discusses in detail the impacts on occupational and public health and safety from the basic implementation of Management Alternative 1, even though the analysis concludes that such impacts are very low. Section 4.2.4.2 summarizes the impacts on the other environmental topics. Section 4.2.4.3 discusses the cumulative impacts of the basic implementation of Management Alternative 1 at each candidate management site, and Section 4.2.4.4 addresses the waste management and mitigation measures available under the basic implementation of Management Alternative 1. Later in this chapter, Section 4.10 compares the risks of the basic implementation of Management Alternative 1 to risks of common activities.

4.2.4.1 Occupational and Public Health and Safety

Possible sources of occupational and public radiological exposure from foreign research reactor spent nuclear fuel include: (1) emissions of radioactive material from incident-free operations, (2) incident-free handling activities, and (3) emissions from accident conditions. Foreign research reactor spent nuclear fuel management is not expected to impact occupational and public health and safety. Nonradiological exposures are not likely to occur during construction or operation of foreign research reactor spent nuclear fuel storage facilities. Radiological exposures are presented in individual subsections for emissions-related impacts, handling-related impacts, and accident-related impacts.

Conservative Assumptions and Impacts to the Public of Incident-Free Site Activities

Doses that could be received by the public during incident-free operation of foreign research reactor spent nuclear fuel storage facilities could only be due to emissions of radioactive material that becomes airborne. The public would be too far from the storage facilities to receive any direct exposure. In summary:

- Doses were calculated for the MEI, defined as an individual living at the management site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions released during foreign research reactor spent nuclear fuel transfer from the transportation cask to the storage facility and from foreign research reactor spent nuclear fuel storage.

- Radiological airborne emissions consist of two parts: (1) emissions from gaseous releases during receipt and unloading of the transportation casks; and (2) emissions during the management period. The emissions during receipt and unloading were calculated conservatively assuming one percent of the foreign research reactor spent nuclear fuel would fail during transport and the associated gaseous fission products would be released during the transfer at the management site. DOE and the Department of State also conservatively assumed that unloading the spent nuclear fuel cask in a dry cell would allow all free gaseous fission products to be released to the environment, while unloading in a wet pool would allow 90 percent of the halogens to be retained in the water. Radiological emissions during wet storage were based on historical data at the Receiving Basin for Offsite Fuels (RBOF) at the Savannah River Site. The emissions during incident-free dry storage would be zero because the spent nuclear fuel would be stored in sealed containers. The methodology and conservative assumptions used for the calculation of radiological emissions under the basic implementation of Management Alternative 1 are discussed in detail in Appendix F, Section F.6.

- Doses were calculated separately for each phase of the program at each candidate management site to accommodate the two-phased implementation of the basic implementation of Management Alternative 1. For example, in the case where the Nevada Test Site, the Hanford Site, or the Oak Ridge Reservation is selected as a Phase 2 site, with the Savannah River Site or the Idaho National Engineering Laboratory as a Phase 1 site, doses were calculated at the Savannah River Site or the Idaho National Engineering Laboratory for Phase 1, and at the Hanford Site, Oak Ridge Reservation, or the Nevada Test Site for Phase 2.

- Doses from an operation which combines an existing wet or dry storage facility for spent nuclear fuel receiving and characterization and dry storage casks to enhance storage capacity are bounded by the doses calculated for the existing facility.

- Doses were conservatively calculated for the maximum quantity of foreign research reactor spent nuclear fuel that could be received at each storage site as discussed in Appendix F, Section F.4.

Tables 4-8 through 4-12 summarize the annual emission-related doses to the public and the associated risks for the MEI and population at each site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period. In general, receipt and unloading at wet storage facilities produces lower public risk than at dry storage facilities.

Table 4-8 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Savannah River Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet storage)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet storage)	0.000073	3.7×10^{-11}	0.0046	0.0000023
• New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at:				
• RBOF (wet storage)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet storage)[a]	0.00036	1.8×10^{-10}	0.022	0.000011
• New Dry Storage Facility	0	0	0	0

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Table 4-9 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF[a]/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• Fluorinel Dissolution and Fuel Storage (FAST) (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
• New Dry Storage Facility[b]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at:				
• IFSF[a]/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
• New Dry Storage Facility[b]	0	0	0	0

[a] *Irradiated Fuel Storage Facility*

[b] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Table 4-10 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Hanford Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Fuel Material Examination Facility (FMEF) (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility[a]	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are different from those for FMEF due to the different release height and location.*

Table 4-11 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Table 4-12 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• Engine Maintenance and Disassembly (E-MAD) (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
• New Dry Storage Facility[a]	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0
• New Dry Storage Facility[a]	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for E-MAD.*

Among all the potential foreign research reactor spent nuclear fuel management sites, the maximum estimated annual incident-free public MEI radiological exposure from emissions is 0.09 mrem per year. This exposure would occur at the Oak Ridge Reservation (Table 4-11) during receipt and handling. It is much higher than all other corresponding dose rates in Tables 4-8 through 4-12. The receipt period would be about 3 years, so the total MEI dose would be 0.27 mrem. The associated probability for incurring one LCF would be 1.4 x 10^{-7} for the MEI, which represents less than two chances in ten million of developing a fatal cancer from radiological exposure.

The highest annual incident-free population risk among the Savannah River Site and the Idaho National Engineering Laboratory (Phase 1 sites) is 0.000011 LCF per year (Tables 4-8 and 4-9), which would be due to emissions from L-Reactor Basin at the Savannah River Site. Assuming some foreign research reactor spent nuclear fuel is stored in this basin for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be as high as 0.00014 LCF. The highest annual incident-free population risk from a new dry storage facility at a potential Phase 2 site (Tables 4-8 through 4-12), is 0.000043 LCF per year, which would be due to receipt/unloading at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.00013 LCF. This is higher than any other combination of Phase 2 dry storage annual risks and durations. Adding the Phase 1 and Phase 2 population risks yields 0.00027 LCF for the total population risk to the public living near the sites due to incident-free conditions.

Conservative Assumptions and Impacts to Workers of Incident-Free Site Activities

Workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the management site, transferring foreign research reactor spent nuclear fuel from one facility to another within the management site, or packaging the foreign research reactor spent nuclear fuel for shipment to another management site. Detailed analysis of the potential impacts is given in Appendix F of this EIS. In summary:

- The maximally exposed worker dose estimate is based on the regulatory limit of 5,000 mrem per year for radiation workers at all DOE management sites. DOE and the Department of State conservatively assumed that an individual worker received this dose every year for all 13 years that the handling operations would be in progress. Although this assumption is highly unlikely, the calculated total maximally exposed worker dose is 65,000 mrem and the associated risk is 0.026 LCF. This means that this individual would have a nearly three percent higher chance of incurring an LCF.

- Worker population doses were estimated by considering the type and duration of all operations performed by the workers during the handling of each transportation cask and storage cask as appropriate, including: (1) the number of workers needed, (2) the duration of a specific operation, and (3) the distance between the transportation cask and the operation being performed. Only the workers actually performing the operations receive radiation doses, and thus would have an increased risk of incurring an LCF. If the total radiation dose is received by a small number of workers, each worker would have a higher risk of cancer than if the total dose is received by a large number of workers. The dose rate in the vicinity of the transportation and storage casks assumed for the estimates was based on the conservative methodology presented in Appendix F, Section F.5. As noted in Section F.5, worker population doses associated with dry storage cask design may be higher than those associated with the vault design because of the additional worker activities associated with the handling of the cask that transfers the canistered spent fuel to the concrete structure. The worker population doses reported below for new dry storage conservatively reflect the cask design.

- The number of casks handled at each potential foreign research reactor spent nuclear fuel management site would depend on the number of cask shipments considered under the ground transportation options discussed in Section 2.6.4.1, and the amount of foreign research reactor spent nuclear fuel expected to be transferred between facilities during Phase 2.

Table 4-13 provides a summary of the number of casks that would be handled at each potential foreign research reactor spent nuclear fuel management site under the Centralization Alternative in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and in the current EIS.

Table 4-13 Estimated Number of Shipments to and from Each Potential Foreign Research Reactor Spent Nuclear Fuel Management Site

Candidate Storage Site	*Incoming Shipments*	*Intersite Shipments*	*Outgoing Shipments*	*Total Shipments*
Savannah River Site or Idaho National Engineering Laboratory Phase 1	644[a]	0	161	805
Savannah River Site or Idaho National Engineering Laboratory Phases 1 and 2	837[b]	209	0	1,046
Hanford Site or Oak Ridge Reservation or Nevada Test Site Phase 2	354[c]	0	0	354

[a] *10-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[b] *13-year receipt in foreign research reactor spent nuclear fuel certified casks.*

[c] *161 from near term site using large truck casks and 193 from ports using foreign research reactor spent nuclear fuel certified casks.*

Tables 4-14 through 4-18 present the population doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at each management site. The results do not include shipments in large rail casks.

Table 4-14 Handling-Related Impacts to Workers at the Savannah River Site

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	RBOF/L-Reactor	*New Dry Storage*	*RBOF/L-Reactor*	*New Dry Storage*
Phase 1	250	NA	0.10	NA
Phases 1 and 2	NA	416[a]	NA	0.17[a]

[a] *Cask design*

Table 4-15 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	IFSF[a]/CPP-749	*FAST*	*New Dry Storage*	*IFSF[a]/CPP-749*	*FAST*	*New Dry Storage*
Phase 1	257	250	NA	0.10	0.10	NA
Phases 1 and 2[b]	NA	NA	424[c]	NA	NA	0.17[c]
Phases 1 and 2[d]	NA	NA	416[c]	NA	NA	0.17[c]

[a] *Irradiated Fuel Storage Facility*

[b] *Phase 1 at IFSF/CPP-749*

[c] *Cask design*

[d] *Phase 1 at FAST*

Table 4-16 Handling-Related Impacts to Workers at the Hanford Site

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	FMEF/New Dry Storage	*FMEF/New Dry Storage*
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-17 Handling-Related Impacts to Workers at the Oak Ridge Reservation

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Dry Storage	*New Dry Storage*
Phase 2	266[a]	0.11[a]

[a] *Cask design*

Table 4-18 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	E-MAD	*New Dry Storage*	*E-MAD*	*New Dry Storage*
Phase 2	113	266[a]	0.05	0.11[a]

[a] *Cask design*

According to the above tables, the highest dose to a working crew at a single site would be 424 person-rem at the Idaho National Engineering Laboratory in the analyzed case which assumes that all foreign research reactor spent nuclear fuel is received in the Irradiated Fuel Storage Facility and/or the CPP-749 facility (dry storage) during Phase 1 and is transferred to a new dry storage facility at the Idaho National Engineering Laboratory in Phase 2. The associated number of additional LCF is 0.17. The highest dose to working crews for both phases in more than one site is 523 person-rem: 266 person-rem at one of the 3 Phase 2 sites, plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.21.

Conservative Assumptions and Accident-Related Impacts

An evaluation of hypothetical accidental radioactive material releases at the potential foreign research reactor spent nuclear fuel management sites was performed to assess the impact of possible radiation exposure to individuals and the general population (see also Appendix F, Section F.6). All inputs are site-specific except for the radioactivity release. Site-specific information includes meteorological conditions, population distribution, and food production and consumption rates within 80 km (50 mi) of the management location.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel management facility:

- Worker: An individual located 100 m (330 ft) from the radioactive material release point. (The impact of accidents on close-in workers is not calculated numerically but is discussed qualitatively for each accident at the end of this section.) For elevated release (from a tall stack), the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release.

The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- Maximally Exposed Individual (MEI): A theoretical member of the general public living at the management site boundary receiving the maximum exposure. This individual is conservatively assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the conservative 95th-percentile meteorological condition.

- Nearest Public Access Individual (NPAI): An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the conservative 95th-percentile meteorological condition.

- General population within an 80-km (50-mi) radius of the facility: The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the conservative 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume the contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeded the protective action guidelines developed by the U.S. Environmental Protection Agency (EPA, 1991a). To ensure a consistent and conservative analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those that were analyzed in the Programmatic SNF&INEL Final EIS. The selection of the accidents was based on the following considerations:

(1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent fuel element); and
(3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each management location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a spent nuclear fuel cask, and an aircraft crash with an ensuing fire.

Tables 4-19 through 4-23 present the frequencies and the consequences of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the conservative assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE and the Department of State did not estimate the worker population dose due to accidents.

Table 4-19 Frequency and Consequences of Accidents at the Savannah River Site

		Consequences							
		MEI		*NPAI*		*Population*		*Worker*	
	Frequency (per yr)	*(mrem)*	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	0.24	1.2×10^{-7}	0.068	3.4×10^{-8}	9.2	0.0046	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.018	9.0×10^{-9}	0.00034	1.7×10^{-10}	0.55	0.00028	0.28	1.1×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	40	0.00002	0.29	1.5×10^{-7}	1300	0.65	120	0.000048
Wet Storage Accidents - RBOF									
Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2.0×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
Accidental Criticality	0.0031	130	0.000065	44	0.000022	4,800	2.4	16,000	0.0064
Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
Wet Storage Accidents - L-Reactor Basin									
Spent Nuclear Fuel Assembly Breach	0.16	0.0093	4.7×10^{-9}	0.00097	4.9×10^{-10}	0.14	0.00007	0.11	4.4×10^{-8}
Accidental Criticality	0.0031	170	0.000085	120	0.000060	3,000	1.5	14,000	0.0056
Aircraft Crash	1×10^{-6}	4.2	0.0000021	2.6	0.0000013	93	0.047	70	0.000028

[a] *New Dry Storage Facility*

Table 4-20 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

		Consequences							
		MEI		*NPAI*		*Population*		*Worker*	
	Frequency (per yr)	*(mrem)*	*(LCF)*	*(mrem)*	*(LCF)*	*(person-rem)*	*(LCF)*	*(mrem)*	*(LCF)*
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.3	6.5×10^{-7}	0.67	3.4×10^{-7}	15	0.0075	28	0.000011
Dropped Spent Nuclear Fuel Cask	0.0001	0.074	3.7×10^{-8}	0.0033	1.7×10^{-9}	0.83	0.00042	0.12	4.8×10^{-8}
Aircraft Crash w/Fire	1×10^{-6}	180	0.00009	2.9	0.0000015	2,000	1.0	120	0.000048
Wet Storage Accidents									
Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8.0×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1800	0.00072
Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016

[a] *New Dry Storage Facility at IFSF/CPP-749*

Table 4-21 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	3.0	0.0000015	0.57	2.9×10^{-7}	42	0.021	50	0.000020
Dropped Spent Nuclear Fuel Cask	0.0001	0.26	1.3×10^{-7}	0.0085	4.3×10^{-9}	3.0	0.0015	0.22	8.8×10^{-8}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF									
Spent Nuclear Fuel Assembly Breach[c]	0.16	4.7	0.0000024	2.1	0.0000011	46	0.023	0.99	4.0×10^{-7}
Dropped Spent Nuclear Fuel Cask[c]	0.0001	0.2	1.0×10^{-7}	0.032	1.6×10^{-8}	3.2	0.0016	0.0049	2.0×10^{-9}
Aircraft Crash w/Fire[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack, so workers would receive low doses.*

NA = Not applicable

Table 4-22 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	22	0.000011	42	0.000021	55	0.028	140	0.000056
Dropped Spent Nuclear Fuel Cask	0.0001	1.4	7.0×10^{-7}	0.18	9.0×10^{-8}	15	0.0075	0.61	2.4×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	2300	0.0012	180	0.000090	2900	1.5	610	0.00024

[a] *New Dry Storage Facility*

Table 4-23 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Dry Storage Accidents[a]									
Spent Nuclear Fuel Assembly Breach	0.16	1.7	8.5×10^{-7}	0.31	1.6×10^{-7}	1.5	0.00075	20	0.0000080
Dropped Spent Nuclear Fuel Cask	0.0001	0.11	5.5×10^{-8}	0.0014	7.0×10^{-10}	0.40	0.00020	0.089	3.6×10^{-8}
Aircraft Crash w/Fire	1×10^{-6}	180	0.000090	1.2	6.0×10^{-7}	250	0.13	87	0.000035

[a] *E-MAD and New Dry Storage Facility*

The analyses were performed for a generic dry storage at the five potential foreign research reactor spent nuclear fuel management sites, as well as for site-specific locations (i.e., FMEF at the Hanford Site, E-MAD at the Nevada Test Site, L-Reactor Basin and RBOF at the Savannah River Site).

Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. These annual risk estimates are presented in Tables 4-24 through 4-28.

Table 4-24 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.9×10^{-8}	5.5×10^{-9}	0.00075	0.0000018
Dropped Spent Nuclear Fuel Cask	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	1.1×10^{-11}
Aircraft Crash w/Fire	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.8×10^{-11}
Wet Storage Accidents at RBOF				
Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
Accidental Criticality	2.0×10^{-7}	7.0×10^{-8}	0.0074	0.000020
Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
Wet Storage Accidents at L-Reactor Basin				
Spent Nuclear Fuel Assembly Breach	7.4×10^{-10}	8.0×10^{-11}	0.000011	7.1×10^{-9}
Accidental Criticality	2.6×10^{-7}	1.9×10^{-7}	0.0047	0.000017
Aircraft Crash	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	2.8×10^{-11}

[a] *New Dry Storage Facility.*

Table 4-25 Annual Risks of Accidents at the Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
Dropped Spent Nuclear Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *FAST Facility*

The highest annual MEI or NPAI risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 2.6×10^{-7} LCF per year, which is the annual risk to the MEI from accidental criticality at L-Reactor Basin. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1 plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this MEI risk would be 0.0000034 LCF. The highest annual MEI or NPAI risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0000034 LCF per year, which is the annual risk to the NPAI from an assembly breach accident during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this MEI/NPAI risk would be 0.000010 LCF. This is higher than any other

Table 4-26 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
Dropped Spent Nuclear Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
Spent Nuclear Fuel Assembly Breach[c]	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
Dropped Spent Nuclear Fuel Cask[c]	8.0×10^{-12}	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
Aircraft Crash with Fire[b]	NA	NA	NA	NA

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

[c] *Emissions would be released through a tall stack.*

NA = Not applicable

Table 4-27 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
Aircraft Crash w\Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table 4-28 Annual Risks of Accidents at the Nevada Test Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
Spent Nuclear Fuel Assembly Breach	1.4×10^{-7}	2.5×10^{-8}	0.00012	0.0000013
Dropped Spent Nuclear Fuel Cask	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	3.6×10^{-12}
Aircraft Crash w\Fire	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	3.5×10^{-11}

[a] *E-MAD and New Dry Storage Facility*

combination of Phase 2 annual accident risks and associated durations. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.000010 LCF for the maximum MEI risk due to accidents. This means that the MEI has one chance in one hundred thousand of incurring an LCF due to accidents.

The highest annual population risk among the potential Phase 1 sites (Tables 4-24 and 4-25) is 0.0074 LCF per year, which is the annual population risk from an accidental criticality at RBOF. Assuming some foreign research reactor spent nuclear fuel is stored in RBOF for the entire 10 years of Phase 1, plus 3 years to transfer it to a Phase 2 site, the Phase 1 component of this population risk would be 0.096 LCF. The highest annual population risk due to dry storage facility accidents at the potential Phase 2 sites (Tables 4-24 through 4-28) is 0.0044 LCF per year, which is the annual risk to the public from assembly breach accidents during handling at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be received at the Oak Ridge Reservation for as long as 3 years, the Phase 2 component of this population risk would be 0.013 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations. Adding the Phase 1 and Phase 2 population risks yields 0.11 LCF for the total population risk due to accidents.

Impacts of Accidents on Close-in Workers

An evaluation has been made of the impacts to close-in workers involved in spent fuel handling and management operations. This evaluation focuses on the radiological consequences of the accident. Clearly, a limited number of fatalities could occur which would be related to spent nuclear fuel handling only in an indirect or secondary manner (e.g., the worker who happened to be in the facility might be killed due to an aircraft crash).

Wet Storage Accidents

Fuel Assembly Breach in Wet Storage: No fatalities of nearby workers would be expected due to radiological consequences. This is because the release of the radionuclides would be underwater. Attenuation by the water would occur for most of the release products; release of the noble gases from the pool would, however, cause a direct radiation exposure to workers in the area. If radiation is released from the surface of the water pool, radiation alarms would sound, prompting evacuation of nearby workers.

Dropped Fuel Cask in Wet Storage: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask.

Accidental Criticality: The accidental criticality could occur at a minimum of 3 m (10 ft) underwater. Based on the shielding provided by the water pool, it is likely that no fatalities would occur. Nearby workers would likely receive appreciable radiation exposures.

Aircraft Crash into the Water Pool: No fatalities to nearby workers would be expected due to radiological consequences. An aircraft crash into the water pool would prompt nearby workers not affected by the crash to evacuate the area immediately. The release of radiation products would be underwater, allowing sufficient time for evacuation before radiation products would reach the surface.

Dry Storage Accidents

Fuel Assembly Breach: Cropping of the fuel assembly would occur in a dry cell. Any release that would occur due to inadvertent cutting into the fuel would be confined to the storage cell, where the exhaust is away from the workers. No fatalities to nearby workers would be expected due to this scenario.

Dropped Fuel Cask: No fatalities would be expected due to the radiological consequences of this accident. The operation of the crane is done using remote controls, so workers are not likely to be in the direct vicinity of the dropped cask. In addition, the workers would promptly leave the area.

Aircraft Crash with Fire: If an aircraft crashes into a dry storage facility and catches fire, large amounts of radioactive material could be released into the atmosphere. If the facility is occupied at the time of the crash, any surviving workers could receive a substantial radiation dose from the released radioactive material.

Secondary Impacts of Accidents

Impacts of accidents on resources other than human health and safety (secondary impacts), have been addressed in Section F.4 for each management site. The general conclusion is that no measurable secondary impacts to land uses, cultural resources, water quality, ecological resources, national defense, or local economies are expected from the postulated accidents involving foreign research reactor spent nuclear fuel at the management sites.

4.2.4.2 Topics Not Discussed in Detail

This section summarizes the potential impacts for the environmental topics not covered in Section 4.2.4.1, namely land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, noise, utilities and energy, and waste management. The detailed analysis of these topics presented in Appendix F, Section F.4 showed that none of these topics clearly differentiated among the potential foreign research reactor spent nuclear fuel management sites nor had any major environmental impact. The discussion of each topic generally concentrates on management sites and alternatives that have the largest estimated impacts, and demonstrates that the environmental impacts for that topic are not of sufficient magnitude to be given strong consideration in the decision making process.

4.2.4.2.1 Land Use

The basic implementation of Management Alternative 1 would only result in minor land use impacts at any of the potential foreign research reactor spent nuclear fuel management sites. The largest land use impact would be 16 ha (40 acres) at the Oak Ridge Reservation to construct a new dry storage facility. This represents less than 0.1 percent of the total size of the Oak Ridge Reservation. A description of the land use impacts at the other potential foreign research reactor spent nuclear fuel management sites is contained in Appendix F.4. For all of the potential foreign research reactor spent nuclear fuel management sites, new foreign research reactor spent nuclear fuel storage facilities would be built on land previously disturbed or designated for industrial use. No additional land outside of the existing sites would be required for foreign research reactor spent nuclear fuel management. It should be noted that land use and other environmental impacts associated with the construction activities would be minimal, under the implementation alternatives that use refurbishment of existing facilities for interim storage (i.e., BNFP at the Savannah River Site and E-MAD at the Nevada Testing Site). All environmental impacts from the refurbishment and operation of these facilities would be bounded by the impacts associated with the construction and operation of new generic storage facilities. Land use impacts are discussed in more detail in Appendix F, Section F.4.

4.2.4.2.2 Socioeconomics

The basic implementation of Management Alternative 1 would only result in minor socioeconomic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Socioeconomic impacts are defined for purposes of this analysis in terms of direct effects, which include changes in site employment and expenditures from foreign research reactor spent nuclear fuel-related construction and operation and indirect effects, such as changes that result from regional purchases, nonpayroll expenditures, and payroll spending by site employees.

No construction personnel would be needed for existing facilities, and not more than 240 workers per year (peak) would be needed to build a new dry storage facility. The annual staffing requirements for operations would be about 30 and 8 full-time employees during receipt and storage, respectively, for a new dry storage facility. This would represent 0.15 to 0.9 percent of the existing work force at any of the potential foreign research reactor spent nuclear fuel management sites. No new hiring would be expected because most positions would be filled by reassignments of the existing work force. Even if all operational positions were filled by new hires, this would represent about an even smaller increase in regional employment. The secondary effects would be even lower.

4.2.4.2.3 Cultural Resources

The basic implementation of Management Alternative 1 would only result in minor cultural impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Cultural, archaeological, historic, and architectural resources are defined as prehistoric and historic sites, districts, structures, and evidence of human use that are considered to be important to a culture, subculture, or a community for scientific, traditional, religious, or other reasons.

Although most of the potential foreign research reactor spent nuclear fuel management sites contain areas of archaeological, cultural, or historical interest, little or no direct impacts on cultural resources would be expected because of the location of the foreign research reactor spent nuclear fuel storage facilities. Specific site surveys have not been completed; however, based on existing information, no known cultural resources would be affected by construction or operation of foreign research reactor spent nuclear fuel facilities. Prior to construction, specific site surveys would be conducted. In the event that cultural resources were encountered during construction, the State Historic Preservation Officer would be contacted immediately. Similarly, Tribal leaders would be notified if any Native American resources were found.

4.2.4.2.4 Aesthetic and Scenic Resources

The basic implementation of Management Alternative 1 would only result in minor impacts to aesthetic and scenic resources at any of the potential foreign research reactor spent nuclear fuel management sites. Foreign research reactor spent nuclear fuel storage facilities would be located far from public view in areas previously disturbed or designated for industrial use. Construction activities would generate fugitive dust that could temporarily affect visibility. However, best management practices would be implemented to minimize such conditions. Furthermore, facility operations would not produce emissions that would adversely impact visibility.

4.2.4.2.5 Geology

The basic implementation of Management Alternative 1 would only result in minor geologic impacts at any of the potential foreign research reactor spent nuclear fuel management sites. Except for the potential existence of gold, tungsten, and molybdenum at the Nevada Test Site, geologic resources consist only of surficial sand, gravel, or clay deposits, all of which have low economic value. Construction activities would disturb these surface deposits, but because of the large volume of these materials on the potential foreign research reactor spent nuclear fuel management sites, the impact would be expected to be small.

4.2.4.2.6 Air Quality

The basic implementation of Management Alternative 1 would only result in minor impacts on air quality at any of the potential foreign research reactor spent nuclear fuel management sites. The projected emissions from foreign research reactor spent nuclear fuel storage at the potential management sites would not contribute to Federal or State nonattainment standards. Construction activities would be expected to cause only temporary, minor increases in fugitive dust emissions, but the use of standard dust suppression techniques would be expected to mitigate this problem. Particulate emissions could temporarily affect visibility in localized areas, but would not adversely affect Federal or State attainment standards.

4.2.4.2.7 Water Quality

The basic implementation of Management Alternative 1 would have only minor impacts on water resources at the potential foreign research reactor spent nuclear fuel management sites. Water consumption during construction would require very small amounts of water when compared to daily water usage at the potential management sites.

During operations, the greatest amount of water consumed annually would be about 2.1 million liters (550,000 gal) per year. This amount represents no more than 0.2 percent of the annual water consumption at any of the potential foreign research reactor spent nuclear fuel management sites. At the Nevada Test Site, where available water is limited, a cumulative water supply impact could be important from activities other than foreign research reactor spent nuclear fuel management, but the foreign research reactor spent nuclear fuel management contribution would be very small. Further study of the Ash Meadows sub-basin would be required to specify the exact impact on aquifer yield and integrity.

Under normal operations there would be no direct discharge of effluent to ground or surface waters from a new dry storage facility.

4.2.4.2.8 Ecology

The basic implementation of Management Alternative 1 would only result in minor ecological impacts at the potential foreign research reactor spent nuclear fuel management sites. Under any construction of new facilities, individual or small populations of some wildlife species could be disturbed, displaced, or destroyed. However, the size of the areas affected would be small in relation to the size of the potential foreign research reactor spent nuclear fuel management sites and the size of remaining natural habitats. The type of habitats affected could vary but would be typical of the regional area in which the foreign research reactor spent nuclear fuel storage facility is located. For this reason, any such habitat losses would probably not affect any threatened or endangered species or critical habitats in the area. Habitat fragmentation is not expected because new storage facilities would be constructed on land that has been previously disturbed or designated for industrial purposes. Mitigation plans would be developed in consultation with the appropriate agencies if any threatened or endangered species were identified.

DOE has begun or has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction site of foreign research reactor spent nuclear fuel storage facilities at the five potential sites, as required by the Endangered Species Act.

4.2.4.2.9 Noise

The basic implementation of Management Alternative 1 would only result in minor noise impacts at the potential foreign research reactor spent nuclear fuel management sites. Construction activities would generate noise levels consistent with light industrial activity. Based on existing studies these noises would not be expected to propagate offsite at levels that would affect the general population. Noises generated during operations would be less than those during construction.

4.2.4.2.10 Materials, Utilities, and Energy

The basic implementation of Management Alternative 1 would only result in minor impacts on materials, utilities, and energy at the potential foreign research reactor spent nuclear fuel management sites. For existing facilities, incremental increases in materials, utilities, and energy would be very small. New dry storage facilities would result in increased demands on water, power, and sewage. The increased water usage during construction would add no more than 0.2 percent to existing sitewide levels. Increased annual electricity requirements would be about 800 to 1,000 megawatt hours per year and the increased sewage generation would be no more than 1.59 million liters per year (420,000 gal per year), which is less than one percent above existing sitewide levels. At the Nevada Test Site, a central sewage treatment system would have to be constructed for spent nuclear fuel management activities, which would include the foreign research reactor spent nuclear fuel storage facilities. However, all other existing system capacities could manage the estimated increases for materials, utilities, and energy.

4.2.4.2.11 Waste Management

The basic implementation of Management Alternative 1 would only result in minor waste management impacts at the potential foreign research reactor spent nuclear fuel management sites. At all potential management sites the amount of waste generated from foreign research reactor spent nuclear fuel storage is very small when compared to the annual waste projection for each site.

4.2.4.3 Key Cumulative Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

All of the potential foreign research reactor spent nuclear fuel management sites contain facilities unrelated to foreign research reactor spent nuclear fuel that may continue to operate throughout the foreign research reactor spent nuclear fuel program (approximately 40 years). Impacts from both construction and operation of foreign research reactor spent nuclear fuel facilities would be cumulative with the impacts of existing and planned facilities or actions such as environmental restoration and waste management activities unrelated to foreign research reactor spent nuclear fuel and impacts from the management of DOE's spent nuclear fuel inventory.

This section compares the impacts of the basic implementation of Management Alternative 1 and of the implementation alternatives presented in Section 4.3 to the cumulative impacts at each site. The site-specific cumulative impacts are discussed in more detail in Appendix F.

4.2.4.3.1 Key Cumulative Impacts at the Savannah River Site

Table 4-29 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Savannah River Site, including:

Table 4-29 Key Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	FRR SNF Receipt and Storage Contribution	FRR SNF Receipt and Chemical Separation Contribution	Current Activities[a]	Other Activities[b]	Cumulative Impact
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00036	0.66	0.25	4.1	5.0
LCF (per year)	1.8×10^{-10}	3.3×10^{-7}	1.25×10^{-7}	0.000002	0.0000025
• Population Dose (person-rem/yr)	0.022	27	9.1	295	331
LCF (per year)	0.000011	0.014	0.0045	0.15	0.17
• Worker Collective Dose (person-rem/yr)	10[c]	21	263	1,418	1700
LCF (per year)	0.004	0.0084	0.10	0.57	0.68
Waste Generation:					
• High-Level (canisters/yr)	0	6.5	(d)	190[e]	190[e]
• Saltstone (m^3/yr)	0	370	(d)	60,000	60,000
• Transuranic (m^3/yr)	0	0	(d)	1,038	1,038
• Mixed/Hazardous (m^3/yr)	0	8	(d)	2,561	2,569
• Low-Level (m^3/yr)	22	5,700	(d)	35,600	41,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

a *Based on 1993 site data*

b *Other activities include: interim management of nuclear materials, spent nuclear fuel management, Vogtle plant operation, defense waste processing facility, stabilization of plutonium-solutions, site-wide waste management activities, tritium accelerator facility, disposition of surplus HEU, storage and disposition of weapons-usable fissile materials, and the stockpile stewardship and management program activities.*

c *The dose is due to the handling of the FRR SNF during receipt and transfer between facility, averaged over 40 years.*

d *Included in "other activities"*

e *Expected Defense Waste Processing Facility canister production rate (DOE, 1995b).*

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.
- The implementation of the preferred alternative in the Management of Nuclear Materials EIS.
- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF & INEL Final EIS.
- Completion of the construction and operation of the Defense Waste Processing Facility.
- Processing of F-Canyon plutonium solutions to metal.
- Treatment and minimization of radioactive and hazardous wastes at the site as identified in the Savannah River Site Waste Management Final EIS.
- Construction of an accelerator for tritium production at the Savannah River Site, along with associated support facilities.
- Disposition of Surplus Highly Enriched Uranium at the site.
- Storage and Disposition of Weapons-Usable Fissile Materials.

- Stockpile Stewardship and Management Program.
- Current Savannah River Site projects (based on 1993 data).

Table 4-29 also shows the impacts of receipt and near-term chemical separation at the Savannah River Site, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-29 show that the contribution of foreign research reactor spent nuclear fuel to the cumulative impacts at the Savannah River Site would be minimal.

4.2.4.3.2 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Table 4-30 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Idaho National Engineering Laboratory, including the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS, and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Table 4-30 Key Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	*FRR SNF Receipt and Storage Contribution*	*FRR SNF Receipt and Chemical Separation Contribution*	*Current Activities*[a]	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:					
• MEI Dose (mrem/yr)	0.00056	0.048	0.056	0.0057	0.11
LCF (per year)	2.8×10^{-10}	2.4×10^{-8}	2.8×10^{-8}	2.8×10^{-9}	5.5×10^{-8}
• Population Dose (person-rem/yr)	0.0045	0.39	0.34	32	33
LCF (per year)	2.3×10^{-6}	0.00020	0.00017	0.016	0.016
• Worker Collective Dose (person-rem/yr)	10[b]	18	30	344	392
LCF (per year)	0.004	0.0072	0.012	0.137	0.16
Waste Generation:					
• High-Level (canisters/yr)	0	7.5	0	327[c]	327[c]
• Grout (m^3/yr)	0	167	0	875[d]	875[d]
• Transuranic (m^3/yr)	0	0	712	46	758
• Mixed/Hazardous (m^3/yr)	0	8	243	8	259
• Low-Level (m^3/yr)	22	5,700	4,795	2,800	13,300

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a tritium accelerator facility, and the disposition of weapons-usable fissile materials.*

[b] *The dose is due to the handling of FRR SNF during receipt and transfer, averaged over 40 years.*

[c] *Assumed canister production rate (DOE, 1995b).*

[d] *Design capacity of the proposed Waste Immobilization Facility, which is not funded.*

Table 4-30 also shows the impacts of receipt and near-term chemical separation at the Idaho National Engineering Laboratory, from Implementation Alternative 6 of Management Alternative 1 in Section 4.3.6. These impacts are sufficiently distinct from those of the other alternatives that they are presented separately. These impacts would occur only while the chemical separation facilities are operating.

The results in Table 4-30 show that the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

4.2.4.3.3 Key Cumulative Impacts at the Hanford Site

Table 4-31 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Hanford Site, including those discussed in the Programmatic SNF&INEL Final EIS, the Management of Spent Nuclear Fuel from the K Basins Draft EIS, and the Safe Interim Storage of Hanford Tank Wastes Final EIS.

Table 4-31 Key Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00025	0.0036	0.0036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population Dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective Dose (person-rem/yr)	8.9[b]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Waste Generation:			
• High-Level (canisters/yr)	0	320[c]	320[c]
• Transuranic (m^3/yr)	0	240	240
• Mixed/Hazardous (m^3/yr)	0	402	402
• Low-Level (m^3/yr)	22	33,310	33,332

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a Laser Interferometer Gravitational-Wave Observatory, decommissioning of unused facilities, site restoration activities, interim storage and tank wastes, management of spent nuclear fuel from the K basins, and current activities.*

[b] *The dose is due to the handling of FRR SNF during receipt, averaged over 30 years.*

[c] *Assumed canister production rate (DOE, 1995b).*

The results in Table 4-31 show that the contribution from management of foreign research reactor spent nuclear fuel to the cumulative impacts at the Hanford Site would be minimal.

4.2.4.3.4 Key Cumulative Impacts at the Oak Ridge Reservation

Table 4-32 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Oak Ridge Reservation, including those discussed in the programmatic SNF&INEL Final EIS, the Tritium Supply and Recycling Final EIS, and the Disposition of Surplus Highly Enriched Uranium Draft EIS. Other activities considered for the Oak Ridge Reservation which could affect the site environment have not been determined sufficiently at this time to allow impact evaluation. They include activities associated with the waste management at the site, storage and disposition of weapons-usable fissile materials, and stockpile stewardship and management program.

Table 4-32 Key Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.09	15.5	15.6
LCF (per year)	4.5×10^{-8}	0.0000077	0.0000078
• Population Dose (person-rem/yr)	0.085	94.5	94.6
LCF (per year)	0.000043	0.047	0.047
• Worker Collective Dose (person-rem/yr)	8.9[b]	261.3	270.2
LCF (per year)	0.0036	0.104	0.108
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	119,411	119,411
• Low-Level (m^3/yr)	22	34,989	35,011

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of the Expended Core Facility, the construction and operation of the Advanced Neutron Source Facility, construction and operation of a Tritium production facility, and surplus highly-enriched uranium management activities at the site.*

[b] *The dose is due to the handling of FRR SNF during receipt, averaged over 30 years.*

The results in Table 4-32 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Oak Ridge Reservation would be minimal.

4.2.4.3.5 Key Cumulative Impacts at the Nevada Test Site

Table 4-33 presents the key cumulative impacts from ongoing actions and reasonably foreseeable actions at the Nevada Test Site, including those discussed in the Programmatic SNF&INEL Final EIS and the Tritium Supply and Recycling Final EIS. The Programmatic SNF&INEL Final EIS includes the quantitative impacts from a proposed Expended Core Facility at the Site. The Nevada Test Site is also considered in the storage and disposition of weapons-usable fissile materials program which could affect the site environment. The impacts from this program have not been determined sufficiently at this time to allow impact evaluation.

The results in Table 4-33 show that the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Nevada Test Site would be minimal.

4.2.4.4 Waste Minimization and Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Although environmental impacts at the potential foreign research reactor spent nuclear fuel management sites would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Mitigation measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Appendix F provides details on these issues.

Table 4-33 Key Cumulative Impacts at the Nevada Test Site

Environmental Impact Parameter	FRR SNF Contribution	Other Activities[a]	Cumulative Impact
Occupational and Public Health and Safety:			
• MEI Dose (mrem/yr)	0.00076	0.31	0.31
LCF (per year)	3.8×10^{-10}	1.55×10^{-7}	1.55×10^{-7}
• Population Dose (person-rem/yr)	0.00093	0.095	0.095
LCF (per year)	4.7×10^{-7}	0.00047	0.000047
• Worker Collective Dose (person-rem/yr)	8.9[b]	81	89.9
LCF (per year)	0.0036	0.032	0.035
Waste Generation:			
• High-Level (canisters/yr)	0	0	0
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	252	252
• Low-Level (m^3/yr)	22	44,578	44,600

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include existing activities, DOE-owned spent fuel management activities, construction and operation of an Expended Core Facility, and construction and operation of a tritium production facility.*

[b] *The dose is due to the handling of foreign research reactor spent nuclear fuel during receipt, averaged over 30 years.*

4.2.4.5 Environmental Justice at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Under incident-free foreign research reactor spent nuclear fuel management site activities associated with receipt and storage of the spent nuclear fuel, the dominant radiological impacts would be the exposures received by the site workers in the immediate vicinity of the spent nuclear fuel container. These individuals are principally those working within the spent nuclear fuel storage facility. As discussed in Section 4.2.4.1, under incident-free operating conditions, no radiological fatalities would be expected among radiation workers or the general public.

Section 4.2.4.1 also discusses radiological effects due to accidents for both wet storage and dry storage. As shown in Tables 4-24 through 4-28, the dominant radiological risks due to accidents are estimated to occur during breach of a spent nuclear fuel assembly. No LCF are expected to result from the basic implementation of Management Alternative 1.

Appendix A describes minority populations and low-income households residing near candidate management sites. Table 4-34 summarizes this description. Calculations for incident-free and accident conditions demonstrate that for the general population the impacts would be very low. Minority or low-income populations living near the potential management sites would not be subjected to any greater impacts. Therefore, these populations would not receive disproportionately high and adverse impacts. They would be subject to very low impacts, as would the general population.

Table 4-34 Summary Description of Minority Populations and Low-Income Households Residing Within 80 km (50 mi) of Candidate Management Sites

Candidate Management Site	Total Population	Minority Population	Total Households	Low-Income Households
Savannah River Site	566,823	214,016	197,937	82,930
Idaho National Engineering Laboratory	176,311	15,449	55,109	22,452
Hanford Site	383,934	95,042	136,496	57,667
Oak Ridge Reservation	863,758	53,185	335,589	147,537
Nevada Test Site	12,421	2,005	4,194	2,024

Characterization of the number and location of minority and low-income populations is dependent on how these populations are defined and what assumptions are used in conducting the analysis. As discussed in Appendix A, at the time this Final EIS and the Programmatic SNF&INEL Final EIS were prepared, the Federal Interagency Working Group on Environmental Justice had not issued final guidance on the definitions of minority and low-income populations, or the approach to be used in analyzing environmental justice, as directed by the Executive Order. Final internal DOE guidance on environmental justice has also not been adopted. As a result, both the definitions and assumptions used by and within agencies for conducting environmental justice analyses can vary, and the resulting demographic results can differ on a case-by-case basis. For example, this Final EIS and the Programmatic SNF&INEL Final EIS present demographic characterizations derived from the same United States Census Bureau data base, but these documents used different definitions and assumptions. Several of the same candidate interim spent nuclear fuel management sites were evaluated in both documents. As discussed in Appendix A, variations in these definitions and assumptions led to differences in the characterization of minority and low-income populations surrounding these potential spent nuclear fuel management sites. Nevertheless, although the characterizations differ, the radiological impacts resulting from the proposed action under all alternatives present very low risk to the population as a whole. Therefore, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations, regardless of which set of definitions and assumptions were applied.

Implementation of the proposed action would have extremely low nonradiological effects on the environment at interim management sites, including the social and economic status of the general population, minority populations, and the low-income population surrounding interim management sites. Economic benefits that would result from increased cargo handling, transportation, and storage at interim management sites would be extremely small for the general population or any particular segment of the population residing near interim management sites.

4.2.4.6 Mitigation Measures at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

Based on the analyses of the environmental consequences for each potential foreign research reactor spent nuclear fuel management site included in Section F.4 of Appendix F, no mitigation measures would be necessary since all potential environmental impacts are substantially below acceptable levels or promulgated standards. However, each potential site would follow operation practices that would minimize the impacts in such areas as pollution prevention, cultural and ecological resources, ground and surface water quality, air quality, noise, traffic, operational and public health and safety, and accident prevention and mitigation. Descriptions of these practices are included in Appendix F, under Mitigation Measures for each site.

4.2.5 Short-Term Uses and Long-Term Productivity

Short-term impacts would be those associated with construction and operation of the storage facilities. No land would be used for the marine or ground transportation of foreign research reactor spent nuclear fuel. The use of land at the potential foreign research reactor spent nuclear fuel management sites would be in conformity with the land use policy of each site. The construction of new storage facility would lead to the loss of small acreage of terrestrial habitat. After adoption of an overall strategy for interim storage of all DOE-owned spent nuclear fuel (including spent fuel from foreign research reactors), some of the areas currently used for interim storage of spent nuclear fuel may be released for other productive uses

(DOE, 1995c). Ecological resources would be directly affected at the area of construction by land clearing. These resources would be limited to small mammals, reptiles, and songbirds. Given the small area that would be used, the overall effect would be of limited impacts on local populations and resources.

4.2.6 Irreversible and Irretrievable Commitments of Resources

The only irreversible use of resources during the marine and ground transportation of foreign research reactor spent nuclear fuel would be the use of petroleum fuel. Irreversible and irretrievable commitment to resources associated with management site activities are discussed below.

4.2.6.1 Management Site Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of foreign research reactor spent nuclear fuel management site facilities would involve materials that could not be recovered or recycled, or resources that would be consumed or reduced to unrecoverable forms, including electrical energy, fuel, construction materials, and miscellaneous chemicals. Some construction materials are recyclable. Some of the resources would be irretrievable because of the nature of the commitment or the cost of reclamation. For example, human resources used for the construction and operation of the potential foreign research reactor spent nuclear fuel storage facilities would be irretrievably lost since these resources would be unavailable for use in other work activity areas. On the whole, foreign research reactor spent nuclear fuel management would not be particularly resource-intensive. The quantities of irreversible and irretrievable resources for each site are included in Appendix F, Section F.4.

4.2.6.2 Energy Resources

Under the basic implementation of Management Alternative 1, about 4.6 metric tons (5.1 tons) of highly-enriched uranium would be accepted into the United States. The energy content of this uranium would be equal to about 1.5 million megawatt-days or over 20 million barrels of No. 2 fuel oil if the conversion efficiency were 100 percent.

4.2.7 Impacts of Ultimate Disposition

Ultimate disposition of DOE's spent nuclear fuel, including foreign research reactor spent nuclear fuel, is a high priority. For planning purposes, DOE has determined that its spent nuclear fuel that is not otherwise managed (e.g., chemically separated, with the high-level waste being converted into a vitrified glass for repository disposal) is authorized for disposal in a geologic repository. Decisions regarding the actual disposition of DOE's spent nuclear fuel will follow appropriate review under the National Environmental Policy Act (NEPA).

It is possible that the foreign research reactor spent nuclear fuel could be accepted intact in a geologic repository. If DOE determines that geologic disposal of intact foreign research reactor spent nuclear fuel is possible, then there would be no onsite impacts beyond those associated with storage and packaging of the foreign research reactor spent nuclear fuel.

It is also possible that some form of processing could be necessary to convert the foreign research reactor spent nuclear fuel into a more stable form prior to its ultimate disposal. This processing could be a near-term new treatment technology, conventional chemical separation, or a new treatment technology that is implemented after an interim period of storage. DOE expects that any new treatment technology would produce no greater impacts than historical chemical separation activities. Therefore, the impacts of near-term treatment of the foreign research reactor spent nuclear fuel would be expected to be no greater than the impacts of chemically separating the same material as discussed in Section 4.3.6. If a new treatment technology is implemented after an interim period of storage and technology development, DOE expects that it would provide a substantial improvement over conventional chemical separation.

When disposal space is available, DOE would transport the intact or processed foreign research reactor spent nuclear fuel to a repository. This transportation would produce impacts similar to the ground transportation impacts discussed in Section 4.4.2.3. Handling and emplacement in the repository would produce impacts similar to those due to handling the spent nuclear fuel or processed waste at the DOE site because similar equipment and procedures would be used and the same regulatory limits on radiation doses would apply.

Yucca Mountain is the candidate site for a geologic repository for both spent nuclear fuel and high-level waste. Under the Nuclear Waste Policy Act, Congress found that a national problem had been created by the accumulation of spent nuclear fuel from commercial reactors and the accumulation of high-level waste. The Nuclear Waste Policy Act assigned to DOE the responsibility for managing the disposal of this spent nuclear fuel and high-level waste, specified the siting process, and authorized the construction of one geologic repository. Under the Nuclear Waste Policy Act Amendments Act of 1987, the process for selecting this repository was streamlined, and the Yucca Mountain site in Nevada was selected as the candidate site for a geologic repository.

Because the environmental documentation process for geologic disposal was established by the Nuclear Waste Policy Act, this EIS does not analyze environmental impacts of disposal at Yucca Mountain or alternative locations. After emplacement in a geologic repository, however, DOE expects there would be no more impacts to workers, the public, or the environment because the radioactive material would be effectively isolated.

In the event that a geologic repository were to be delayed, DOE assumed for purposes of this analysis that it would continue to manage the foreign research reactor spent nuclear fuel, or the high-level radioactive waste resulting from the chemical separation or other processing of such spent nuclear fuel, at the management sites until a geologic repository becomes available. The risk associated with this continued management is low and would not exceed the annual risk discussed in Section 4.2.4.1.

4.2.8 Summary of the Impacts of the Basic Implementation of Management Alternative 1

The principal impacts under the basic implementation of Management Alternative 1 would be occupational and public health and safety impacts. These are presented in Table 4-35 in terms of the risk of death due to cancer for each segment of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-35 shows that the greatest radiological risks would occur during ground transport or site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments expose people at highway rest stops for times about equal to the actual driving times, and (3) one individual at the DOE site receives the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-35 Maximum Estimated Radiological Health Impacts of the Basic Implementation of Management Alternative 1

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.071
Accidents	1.4×10^{-11}	0.00028	---
Site Activities			
Incident-Free	0.026	0.00027	0.21
Accidents	0.000010	0.11	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000010	----	----
Total Population Risk			
Incident-Free	----	0.22	0.33
Accidents	----	0.11	----

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.4×10^{-7} LCF.

The highest estimated accident MEI risk is 0.000010 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-35, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.33 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 33 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-35. DOE and the Department of State estimate there could be about a 14 percent chance that a truck driver or member of the public could die in a traffic accident associated with the basic implementation of Management Alternative 1. This death would result from the traffic accident trauma and would be unrelated to the radioactive nature of the cargo.

4.3 Implementation Alternatives of Management Alternative 1

As discussed in Chapter 2, a policy of managing foreign research reactor spent nuclear fuel in the United States could be implemented by various means. These variations on the basic implementation of Management Alternative 1 of the proposed action have been grouped into seven implementation alternatives. This section discusses their policy considerations and environmental impacts. For convenience, the seven implementation alternatives are listed briefly below:

1. Acceptance of amounts of material different from the amount in the basic implementation of Management Alternative 1,

2. Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the period of time in the basic implementation of Management Alternative 1,

3. Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1,

4. Taking title to the foreign research reactor spent nuclear fuel at locations different from the location in the basic implementation of Management Alternative 1,

5. Use of wet storage technology for the interim period instead of dry storage technology as in the basic implementation of Management Alternative 1,

6. Near term conventional chemical separation of the foreign research reactor spent nuclear fuel instead of interim storage as in the basic implementation of Management Alternative 1, and

7. Development and use of a new processing technology instead of interim storage as in the basic implementation of Management Alternative 1.

4.3.1 Implementation Alternative 1: Alternative Amounts of Spent Nuclear Fuel to be Accepted

DOE and the Department of State have evaluated the policy considerations and environmental impacts for different amounts of spent nuclear fuel and target materials under this implementation alternative.

4.3.1.1 Implementation Subalternative 1a: Accept Foreign Research Reactor Spent Nuclear Fuel Only From Developing Nations

Policy Considerations

Under this implementation subalternative, up to 1.9 MTHM and about 5,000 elements of foreign research reactor spent nuclear fuel would be accepted into the United States from developing nations (defined by the World Bank as nations with other-than-high-income economies). Up to about 238 kg (525 lb) of HEU would be removed from international commerce. By excluding developed countries, which generally share our nuclear weapons nonproliferation goals, but do not necessarily share our belief in the necessity for removing HEU from use in civil programs, this subalternative would have adverse consequences for U.S. nuclear weapons nonproliferation policy.

Because the United States has been unable to accept shipments of HEU spent nuclear fuel since 1988, several foreign research reactor operators have run out of storage capacity or face safety and regulatory problems associated with the presence of spent nuclear fuel at their sites. If the United States is unable to accept any near term shipments of spent nuclear fuel from developed countries, some reactor operators will be forced to either shut down their reactors or ship their spent nuclear fuel for reprocessing to the United Kingdom Atomic Energy Authority facility in Dounreay, United Kingdom, which is the only facility currently able and willing to reprocess foreign research reactor spent nuclear fuel. Operators in Belgium and Germany have already sent spent nuclear fuel elements to Dounreay for reprocessing. Since neither Dounreay nor any other facility is currently accepting aluminum-based research reactor spent nuclear fuel containing LEU for reprocessing, the only way a reactor operator can use reprocessing to control his spent nuclear fuel inventory is by using HEU for fuel. This could lead reactor operators to delay or cancel plans to convert to LEU, or, in some cases, to reconvert from LEU to HEU fuels.

The net result of reduced reliance on the United States is that foreign research reactor operators would be compelled to withdraw from the Reduced Enrichment for Research and Test Reactors (RERTR) program and continue operations on the HEU fuel cycle, with its lower costs and enhanced performance. Since the United States is barred from exporting HEU to virtually all foreign research reactors under the Energy Policy Act of 1992, operators would be forced to seek alternative suppliers of HEU, such as Russia and China. This could lead to renewed international commerce in weapons-usable HEU and undermine the U.S. nuclear weapons nonproliferation policy goal of seeking to minimize the civil use of HEU. Further, those countries that participated in the RERTR program considered U.S. acceptance of their spent nuclear fuel as a condition for incurring the substantial costs and technical difficulties of converting to LEU fuels. Failure to accept their spent nuclear fuel would jeopardize the nuclear weapons nonproliferation goals of the RERTR program and the reputation of the United States as a reliable partner in the conduct of international nuclear materials management.

There is another way this subalternative could undercut the RERTR program. The developing countries generally assess their technical capabilities by comparing themselves with the developed states of North America, Western Europe, and Japan. As noted above, one probable result of this subalternative is that more developed states will continue to use HEU-fueled research reactors, due to difficulty in reprocessing LEU spent nuclear fuel. If that happens, developing countries are likely to regard use of HEU-fueled reactors as more advanced and prestigious than LEU-fueled reactors, increasing the demand for such reactors as well as for HEU itself. Again, this would encourage increased stockpiles of HEU in various developed and developing countries, contrary to U.S. nuclear weapons nonproliferation policy.

If some countries are forced to shut down their reactors and thereby forego the medical and scientific benefits of these reactors, such a situation may lead to criticism that the United States is not a dependable nuclear partner. Some countries, including those in the developing world that have characterized the Treaty on the Non-Proliferation of Nuclear Weapons as a discriminatory bargain between the nuclear "haves" and the nonnuclear "have-nots," may be inclined to accuse the United States, fairly or unfairly, of having failed to comply with its Article IV Treaty pledge to facilitate "the fullest possible exchange of equipment, materials and scientific and technological information for the peaceful uses of nuclear energy." Actions that foster such negative perceptions would undoubtedly complicate the conferences which are scheduled to monitor compliance with the Non-Proliferation Treaty, and may complicate United States diplomatic efforts to attain other arms control and nuclear weapons nonproliferation objectives.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation were analyzed in the same manner as the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.008 to 0.009 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts result from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed in the evaluation of the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 1a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 114 mrem per year. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, 23 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest estimated risks due to an accident during marine transport would therefore be 0.00004 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of 1×10^{-10} LCF. This individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 1a, accepting spent nuclear fuel from developing nations only, results in 23 percent of the total number of cask shipments that are required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative range from 0.0008 to 0.003. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports of entry based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 5×10^{-8} to 0.000004 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of shipments, so the MEI risk is reduced to 5×10^{-11} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner for Implementation Subalternative 1a as for the basic implementation of Management Alternative 1. The results are presented in Figures 4-6 through 4-9. Incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.002 to 0.06 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 potential foreign research reactor spent nuclear fuel management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.001 to 0.015. The estimated number of radiation-related LCF for the general population ranged from 0.0006 to 0.045, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0002 to 0.01.

Impacts of Accidents During Ground Transport

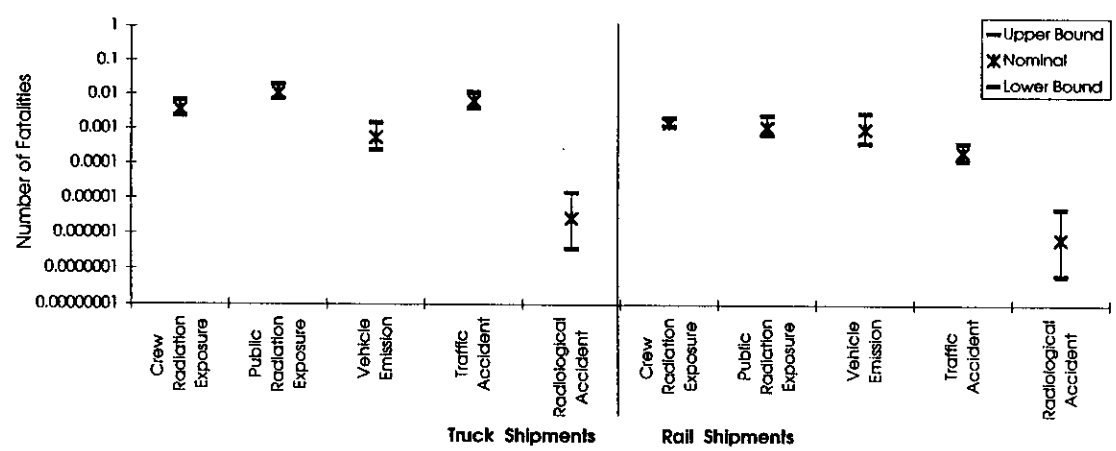
The transportation accident population risks over the entire program are estimated to range from 0.0000001 to 0.00006 LCF from radiation and from 0.0001 to 0.028 traffic fatalities, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 2.7×10^{-12} LCF due to the reduced amount of ground transport.

Management Site Impacts

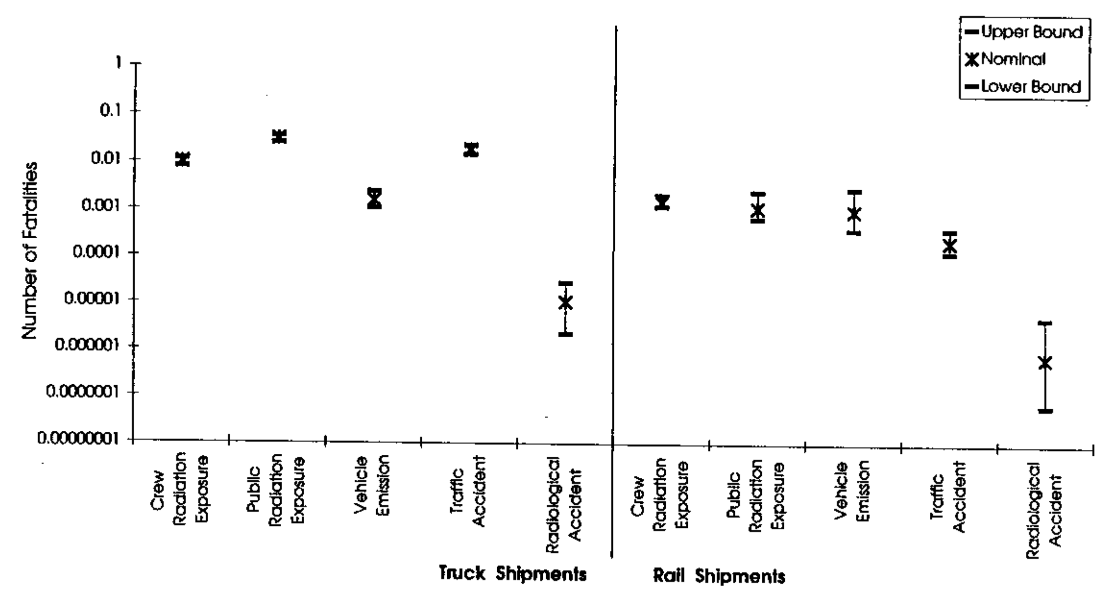
Impacts of Incident-Free Management Site Activities

Impacts of incident-free site activities from Implementation Subalternative 1a are covered by the impacts from the basic implementation of Management Alternative 1. The maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of foreign research reactor spent nuclear fuel involved, and the duration in this subalternative is identical to the basic implementation of Management Alternative 1 (13 years). Thus, the maximally exposed worker dose is conservatively



- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-6 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-7 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative

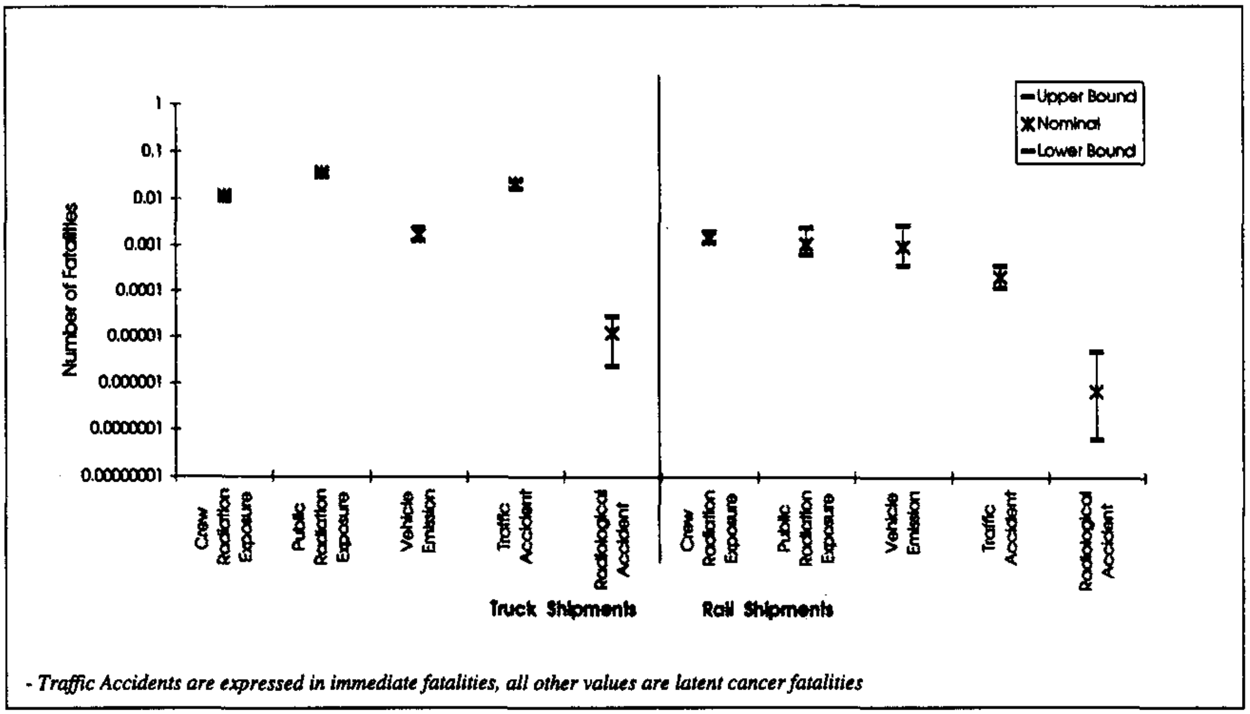


Figure 4-8 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative

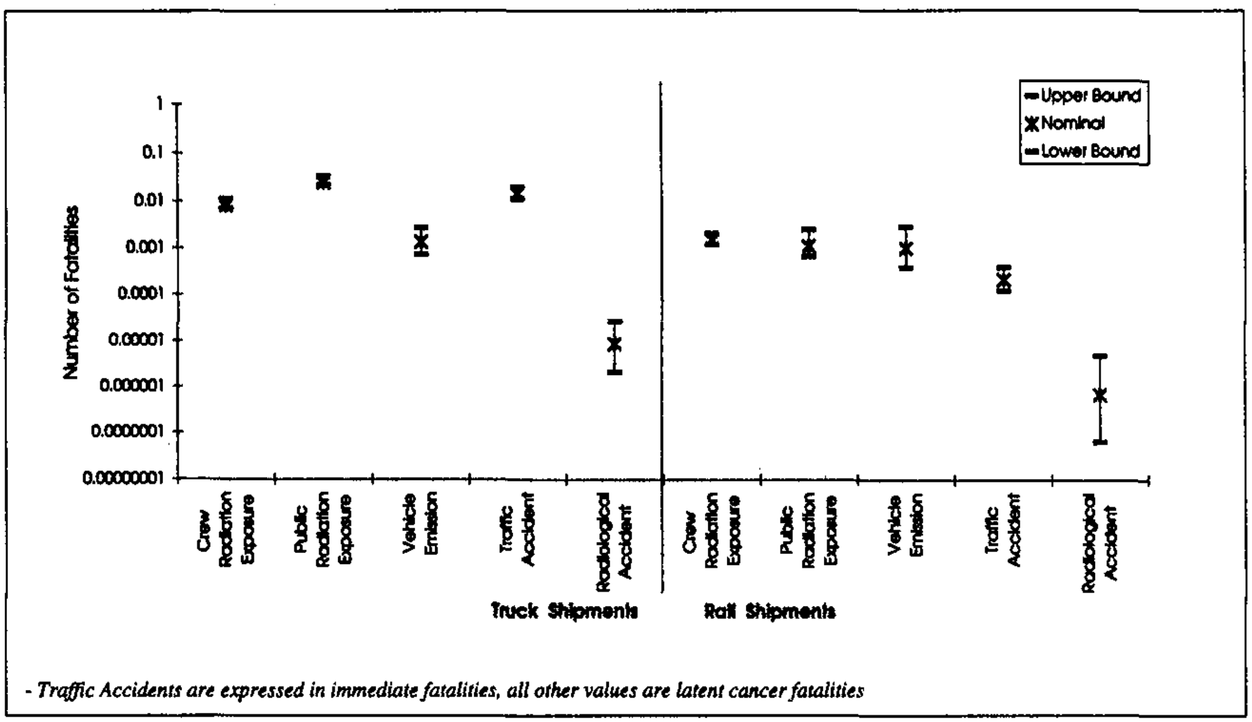


Figure 4-9 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1a and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

assumed to be the same as in the basic implementation of Management Alternative 1. This would produce the maximally exposed worker risk identical to that in the basic implementation of Management Alternative 1 of 0.026 LCF.

The amount of foreign research reactor spent nuclear fuel that would be received and managed is 5,000 elements or approximately 22 percent of the number of elements in the basic implementation of Management Alternative 1. Thus, it is expected that the worker population risks at each management site would be approximately 22 percent of those calculated for the basic implementation of Management Alternative 1. The highest estimate of this risk under the basic implementation of Management Alternative 1 is 0.21 LCF, so the corresponding risk for this subalternative is 0.05, LCF, which is much less than one LCF.

Similarly, some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 22 percent, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by a factor of 22 percent to yield the corresponding risk of 3.1×10^{-8} LCF for this subalternative.

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to storage, so it is not reduced in this subalternative. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.000029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00017 LCF.

Impacts of Accidents Onsite

The highest estimated MEI risk due to accidents in the basic implementation of Management Alternative 1 (0.0000034 LCF) is due to an accidental criticality in RBOF. This MEI risk is greater than any of the potential Phase 2 MEI risks, when those due to receipt/handling are reduced by the factor of 22 percent. Thus, the highest MEI risk due to accidents is 0.0000034 LCF.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. The same facility could be used for the same period of time in this subalternative, so this component of the risk is unchanged. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so it is reduced by the factor of 22 percent to 0.0029 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.099 LCF.

Summary of the Impacts of Implementation Subalternative 1a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-36 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-36 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1a (Developing Nations Only)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.009
Accidents	1×10^{-10}	much less than 0.000004	---
Port Activities			
Incident-Free	0.00052	0	0.003
Accidents	5×10^{-11}	0.000004	---
Ground Transport			
Incident-Free	0.00052	0.045	0.015
Accidents	2.7×10^{-12}	0.00006	---
Site Activities			
Incident-Free	0.026	0.00017	0.05
Accidents	0.0000034	0.099	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.0000034	----	---
Total Population Risk			
Incident-Free	----	0.045	0.077
Accidents	----	0.099	----

Table 4-36 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel produces a dose rate equal to the regulatory limit, (2) truck shipments exposes people at highway rest stops for times about equal to the actual driving times, and (3) one individual at the DOE management site receives the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated indicent-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 3.1×10^{-8} LCF.

The highest estimated accident MEI risk is 0.0000026 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-36, the total incident-free population risk would be 0.045 LCF for the potentially exposed public, and the corresponding risk would be 0.077 LCF for workers. Thus, there would be less than a five percent chance of incurring one additional LCF among the general public, and a 7.7 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-36. There is about a three percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would result from the traffic accident trauma and would be unrelated to the radioactive nature of the cargo.

4.3.1.2 Implementation Subalternative 1b: Accept Only Foreign Research Reactor Spent Nuclear Fuel that Contains HEU

Policy Considerations

Under this implementation subalternative, up to about 4.6 MTHM and 11,200 elements of foreign research reactor spent nuclear fuel would be accepted into the United States. All of this foreign research reactor spent nuclear fuel would contain HEU that was enriched in the United States.

Although this implementation subalternative would remove up to about 4.6 metric tons (5.1 tons) of HEU from international commerce, it almost certainly would result in the end of the RERTR program. As discussed in Chapter 1, the foreign research reactor operators have stated that they would not participate in the RERTR program unless the United States accepts their spent nuclear fuel, including LEU spent nuclear fuel. Otherwise, many research reactor operators would be likely to insist on using HEU fuel in their reactors in the future, which would increase international commerce in HEU. The most likely suppliers of this HEU would be Russia and China. DOE and the Department of State believe that in the long run, this subalternative would be contrary to the broader U.S. policy of nuclear weapons nonproliferation. Therefore, this subalternative is not analyzed in detail for environmental impacts in this EIS.

Summary of the Impacts of Implementation Subalternative 1b

Since the number of elements in this subalternative is about half the number of elements in the basic implementation of Management Alternative 1, the impacts would be roughly half of those calculated for the basic implementation of Management Alternative 1 (see Section 4.2.8).

4.3.1.3. Implementation Subalternative 1c: Accept Target Material in Addition to Foreign Research Reactor Spent Nuclear Fuel

Policy Considerations

This implementation subalternative would entail the shipment to the United States of not only HEU and LEU spent nuclear fuel, but of residual material from the production of molybdenum-99 for medical purposes. Molybdenum-99 is produced by the irradiation of targets in a research reactor. The targets are physically similar to the fuel for foreign research reactors. After being irradiated in a reactor, the targets are dissolved in acid to recover the molybdenum, leaving residual material containing enriched uranium. The United States has supplied HEU to Canada, Belgium, Argentina, and Indonesia for use as targets in the production of medical isotopes. The NRU reactor in Canada produces nearly all radioisotopes used in nuclear medicine in the United States.

This subalternative involves the acceptance of the following amounts of target material from these countries:

Canada	0.525 MTHM
Belgium	0.029 MTHM
Argentina	0.0011 MTHM
Indonesia	0.0014 MTHM
Total	0.5565 MTHM

This total has been rounded up to 0.6 MTHM for the purpose of analysis in this EIS. Under this subalternative, about 216 kg (476 lb) of HEU from target material would be removed from international commerce. This would be in addition to the estimated 4.6 metric tons (5.1 tons) of HEU that would be removed from international commerce under the basic implementation of Management Alternative 1.

Because the residual material contains weapons-usable HEU, there is a strong nuclear weapons nonproliferation rationale for including it in the scope of the management policy. This course of action would be desirable from a nuclear weapons nonproliferation standpoint, since it would leave the United States in control of the disposition of foreign research reactor spent nuclear fuel containing HEU, as well as residuals from the production of molybdenum-99, thereby minimizing the risk that such material might be diverted to a nuclear weapons program. This subalternative removes the most HEU from international civil commerce and provides the most support to U.S. nuclear weapons nonproliferation policy.

Furthermore, this subalternative would give the molybdenum-99 producers an incentive to switch from HEU targets to LEU targets. Appropriate LEU targets are currently under development as part of the RERTR program, and this target material would be accepted under this subalternative subject to the same conditions as the LEU foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1.

The target material may be transported in one of two solid powder forms—as a calcine or an oxide. The calcine form would require about 140 cask shipments, while the oxide form would require about 57 cask shipments. The incident-free and accident risks are different for each form. The calcine material would produce an estimated 2.5 times more incident-free risk, but an estimated 10 times less accident risk than the oxide material. Furthermore, for transporting target material (unlike spent nuclear fuel), the accident risks would be greater than the incident-free risks. Therefore, to estimate conservative radiological risks, DOE and the Department of State assumed the target material would be transported as an oxide powder.

Marine Transport and Port Activities Impacts

The acceptance of target material would cause a very minor change in the marine and port incident-free impacts calculated for the basic implementation of Management Alternative 1. Up to only 7 cask shipments of oxide target material (6 from Belgium and 1[1] from Argentina or Indonesia), excluding the shipments from Canada, are estimated to be needed. This is less than one percent of the marine cask shipments of all foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1. The incident-free impact per shipment is also reduced because the dose rate resulting from a cask loaded with the target material is expected to be much lower than that resulting from a cask loaded with foreign research reactor spent nuclear fuel.

For accident conditions, DOE and the Department of State estimated the risk due to an accident in an east coast port. The risk during marine transport would be much lower than the risk during port activities. The population risk due to accidents during port activities with seven casks of oxide target material is estimated to be 3.2×10^{-9} LCF. This is much lower than the population risk due to accidents with the foreign research reactor spent nuclear fuel.

The MEI risk is estimated to be 2.9×10^{-10} LCF, which is somewhat higher than the corresponding risk for the foreign research reactor spent nuclear fuel, but still very low.

1 *Argentina or Indonesia would not produce enough target material to fill a transportation cask. In all likelihood, the target material from these countries would be shipped along with research reactor spent nuclear fuel elements.*

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation of target material were analyzed in the same manner as for the basic implementation of Management Alternative 1, except that, based on the low activity of the target material, the maximum dose rate at a distance of 2 m (6.6 ft) from the vehicle is estimated to be 0.1 mrem per hour. The risks calculated in this section could be added to those associated with foreign research reactor spent nuclear fuel transport. The incident-free transportation of target material was estimated to result in total latent fatalities that ranged from 0.0002 to 0.003 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew. When the risks of transporting target material are added to the risks of transporting the foreign research reactor spent nuclear fuel, the highest estimate of the population risk is 0.30 LCF.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport target material and combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00007 to 0.00074. The estimated number of radiation-related LCF for the general population ranged from 0.00015 to 0.0023, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.004.

The impacts of transportation related to target material are summarized in Figures 4-10 through 4-13 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

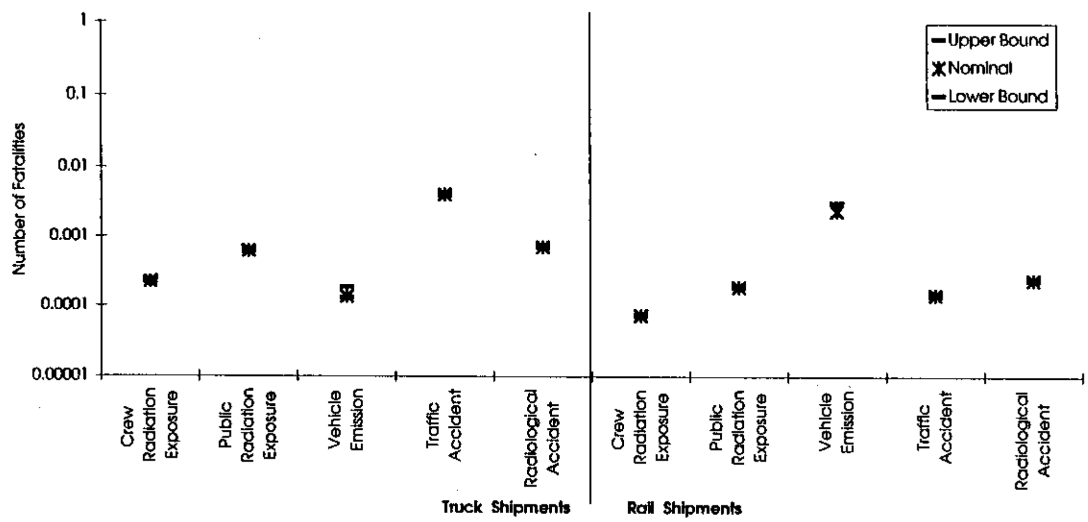
Cumulative transportation accident risks for the target material program are estimated to range from 0.0002 to 0.0054 LCF from radiation and from 0.0001 to 0.013 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation. The highest estimate of the population risk due to accidents involving target material (0.0054 LCF) is higher than the same risk involving foreign research reactor spent nuclear fuel (0.00028 LCF). This difference is due to the physical/chemical forms of the two substances. Adding these two risks together yields the population risk due to accidents under Implementation Subalternative 1c, 0.0057 LCF.

The maximum foreseeable offsite transportation accident involves a cask shipment of powderized target material in a suburban population zone, and the risk is estimated to be 9.3×10^{-11} LCF to the MEI.

The impacts of transportation accidents are summarized in Figures 4-10 through 4-13, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risk of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

Management Site Impacts

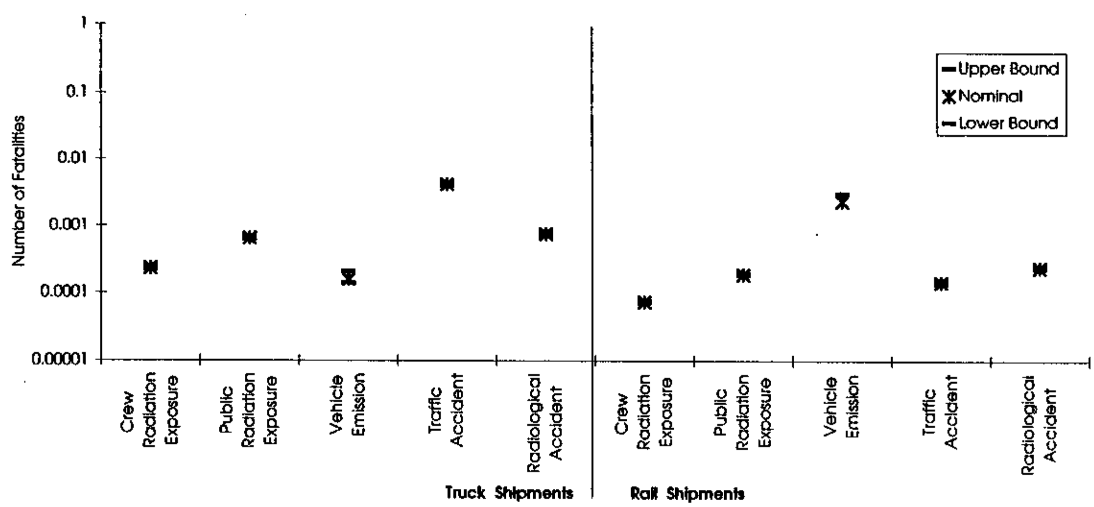
There are two methods of preparing target material for transport. The first is calcining and canning the material with the aluminum included, and the second is to remove the aluminum from the solution, then oxidize and can the residue. Canned material from the first process has similar behavior as that of



- Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives

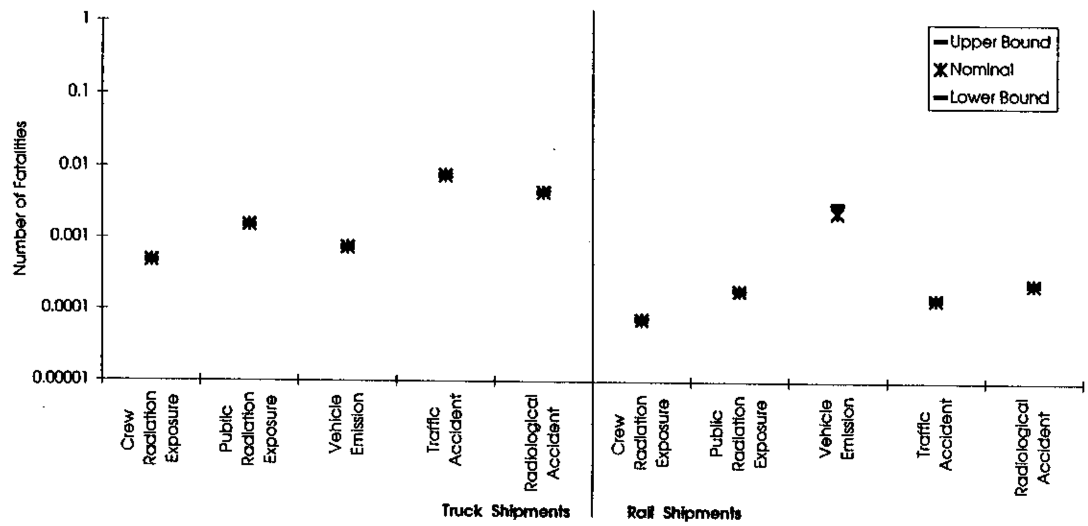
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-10 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Decentralization Alternative



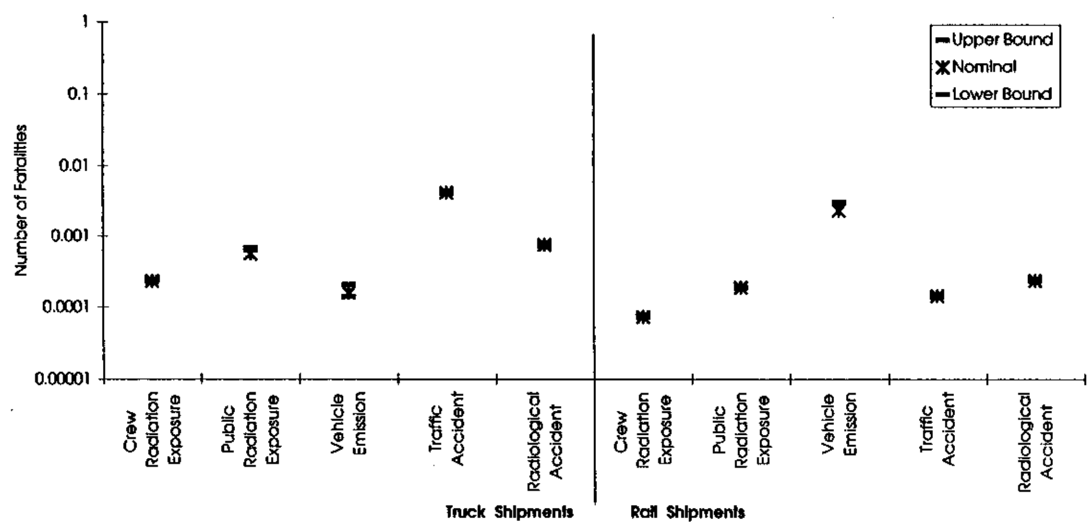
- Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities

Figure 4-11 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-12 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-13 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 1c and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

aluminum-based foreign research reactor spent nuclear fuel containing about 40 g of uranium per can. The second process allows a higher amount of uranium, about 200 g, to be packed in the same size can. Use of the first process would result in 6,750 cans representing approximately 140 cask shipments. The second process would result in 1,350 cans representing approximately 57 cask shipments.

Target material cans would be stored like foreign research reactor spent nuclear fuel elements. The storage space required is a function of volume rather than the nuclear or thermal characteristics of the target material. On average, four cans of target material could be stored in the same space as one foreign research reactor spent nuclear fuel element. Therefore, the maximum storage required for target material (in the 40-gram cans) would be equivalent to 1,700 foreign research reactor spent nuclear fuel elements or approximately 7.4 percent of the space required for the foreign research reactor spent nuclear fuel elements under the basic implementation of Management Alternative 1. The storage facilities analyzed for the basic implementation of Management Alternative 1 include this margin in the sizing.

Impacts of Incident-Free Management Site Activities

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be the same as in the basic implementation of Management Alternative 1.

The collective dose to the crews that would handle the cask shipments would be 70 person-rem, assuming that the cans from 140 cask shipments would be placed in dry storage casks. The associated worker population risk would be 0.03 LCF. Adding this risk to the worker population risk of the basic implementation of Management Alternative 1 yields 0.24 LCF for the total incident-free worker population risk for Implementation Subalternative 1c.

Impacts of Accidents Onsite

The process by which target material is prepared for shipment (i.e., drying and canning of the solutions, see Appendix B, Section B.1.5) releases all gaseous fission products. In addition, the cans do not require any trimming when they arrive at a storage facility. A review of the hypothetical accident scenarios in the basic implementation of Management Alternative 1 indicates that only the aircraft crash with fire accident scenario would be applicable to target material. The cans are never cut, and there are no gaseous fission products, so the foreign research reactor spent nuclear fuel elements breach scenario would not be applicable. In addition, should an aircraft crash into the wet storage pool where the target material is stored, or if an accidental criticality in the pool were to occur, the radioactivity releases would be bounded by those of the spent nuclear fuel analyzed for these accidents. This is because the radioactive inventory per can is very small compared to that in the bounding foreign research reactor spent nuclear fuel.

A scenario involving an aircraft crash into a dry storage facility with an ensuing fire was analyzed for the target material. The scenario assumptions are similar to those described in Appendix F, Section F.6. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target material containing maximum radionuclide inventories, i.e., that of 40 cans of 200 g of uranium per can cooled for at least 3 years.

The frequency of this event is estimated to be 3 percent of the 1×10^{-6} per year used in the accident analysis of the basic implementation of Management Alternative 1. This is because the number of transfer casks involving target material is less than 3 percent of that used for the approximately 22,700 elements in the basic implementation of Management Alternative 1. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be non-foreseeable. Nonetheless, this accident was analyzed and its frequency is set conservatively at 10^{-7} per year. The analytical procedure was the same as that used in the basic implementation of Management Alternative 1.

The highest estimate of the MEI/NPAI accident risk with target material is 2.0 x 10^{-10} LCF, which would occur at the Oak Ridge Reservation (Table F-118, Appendix F). This risk is lower than the highest MEI/NPAI risk in the basic implementation of Management Alternative 1 (0.000010 LCF), so the risk for this subalternative is the same as in the basic implementation of Management Alternative 1. This hypothetical individual would still have one chance in one hundred thousand of incurring an LCF due to an accident on a site.

The highest estimate of the population risk with target material is 1.9 x 10^{-7} LCF, which also would occur at the Oak Ridge Reservation (Table F-118, Appendix F). To obtain the total population risk for this subalternative, this risk must be added to the corresponding risk from the basic implementation of Management Alternative 1 (0.11 LCF). The population risk due to accidents with target material is so small compared to the risk due to the foreign research reactor spent nuclear fuel that it makes essentially no contribution to the population risk for this subalternative. The population risk due to accidents under this subalternative would be the same as that under the basic implementation of Management Alternative 1.

Summary of the Impacts of Implementation Subalternative 1c

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-37 in terms of the risk of death due to cancer during each of the four segments of this subalternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The impacts of the basic implementation of Management Alternative 1 (Table 4-35) are added to the impacts of managing the target material to obtain the impacts of this subalternative. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-37 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.4 x 10^{-7} LCF.

The highest estimated accident MEI risk is 0.000010 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

Table 4-37 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 1c (Target Material)

	Risks (LCF)		
	Maximum Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.072
Accidents	9.3×10^{-11}	0.0057	---
Site Activities			
Incident-Free	0.026	0.00027	0.24
Accidents	0.000010	0.11	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000010	----	----
Total Population Risk			
Incident-Free	----	0.22	0.36
Accidents	----	0.12	----

As shown in Table 4-37, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.36 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 36 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-37. There is about a 15 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.2 Implementation Alternative 2: Alternative Policy Durations

DOE and the Department of State evaluated the impacts for two different policy durations under this implementation alternative: reducing the policy duration to 5 years and continuing the policy for HEU indefinitely.

4.3.2.1 Implementation Subalternative 2a: Five-Year Policy

Policy Considerations

Under this implementation subalternative, DOE would accept up to about 13 MTHM and about 18,800 elements of foreign research reactor spent nuclear fuel. This subalternative would reduce the number of foreign research reactor spent nuclear fuel elements that would be accepted by the United States to about 83 percent of the amount covered by the basic implementation of Management Alternative 1, and would accelerate the time at which the foreign research reactor operators and the governments of their host countries would become responsible for disposal of their own spent nuclear fuel. Up to about 4.1 metric tons (4.5 tons) of HEU would be removed from international commerce, which is about 0.5 metric tons (0.6 tons) less than under the basic implementation of Management Alternative 1.

This subalternative probably would not provide enough time for the foreign countries, especially the developing countries, to make arrangements for alternate means of managing their spent nuclear fuel. This could pressure various foreign research reactor operators to switch their reactors back to HEU fuel. In addition, it would probably, in effect, force many of the foreign research reactors with lifetime cores to shut down prematurely because it would be very difficult for them to find any means to dispose of their foreign research reactor spent nuclear fuel, other than to have DOE accept it.

Marine Transport Impacts

Impacts of Incident-Free Marine Transport

The impacts of incident-free marine transportation in the 5-year acceptance case were analyzed in the same manner as for the basic implementation of Management Alternative 1. The analysis was performed using the dose rates based on the exclusive-use regulatory limit for the shipment of spent nuclear fuel casks. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.025 to 0.028 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels, and would be the same as for vessels analyzed in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Marine Transport

The consequences of the at-sea accidents for Implementation Subalternative 2a are no different than the consequences of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments, the likelihood of such an accident would be reduced. Under this subalternative, approximately 81 percent of the total number of cask shipments required under the basic implementation of Management Alternative 1 would be needed. The highest risk to a human, expressed in terms of peak dose rate, would be 0.00015 mrem per year from the loss of a damaged cask in the deep ocean. Assuming an individual receives this dose for 5 years, the total MEI risk would be about 4×10^{-10} LCF.

Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. Implementation Subalternative 2a results in approximately 81 percent of the total number of cask shipments that are required in the basic implementation of Management Alternative 1. The incident-free impacts of the port activities would be proportionally reduced. The estimated number of LCF associated with this subalternative ranges from 0.0027 to 0.0098. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The estimated maximally exposed worker risk would be 0.00032 LCF.

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. The port accident risks over the entire program are estimated to range from 3×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

The MEI risk would be lower than that of the basic implementation of Management Alternative 1 due to the reduced number of cask shipments. The highest estimated MEI risk is 1.6×10^{-10} LCF.

Ground Transport Impacts

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.010 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of Phase 1 and Phase 2 management sites that created varying shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.005 to 0.064. The estimated number of radiation-related LCF for the general population ranged from 0.005 to 0.20, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.041.

The maximally exposed worker risk would be lower than that in the basic implementation of Management Alternative 1 due to the reduced acceptance period. The highest estimated MEI risk would be 0.00032 LCF.

The impacts of transportation are summarized in Figures 4-14 through 4-17 and are described in more detail in Appendix E.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000003 to 0.00026 LCF from radiation and from 0.001 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.1×10^{-11} LCF.

The impacts of transportation accidents are summarized in Figures 4-14 through 4-17, as described in the previous section, and are described in more detail in Appendix E. These tables can be used to assess the bounded absolute and relative risks of this subalternative under each representative Programmatic SNF&INEL Final EIS alternative.

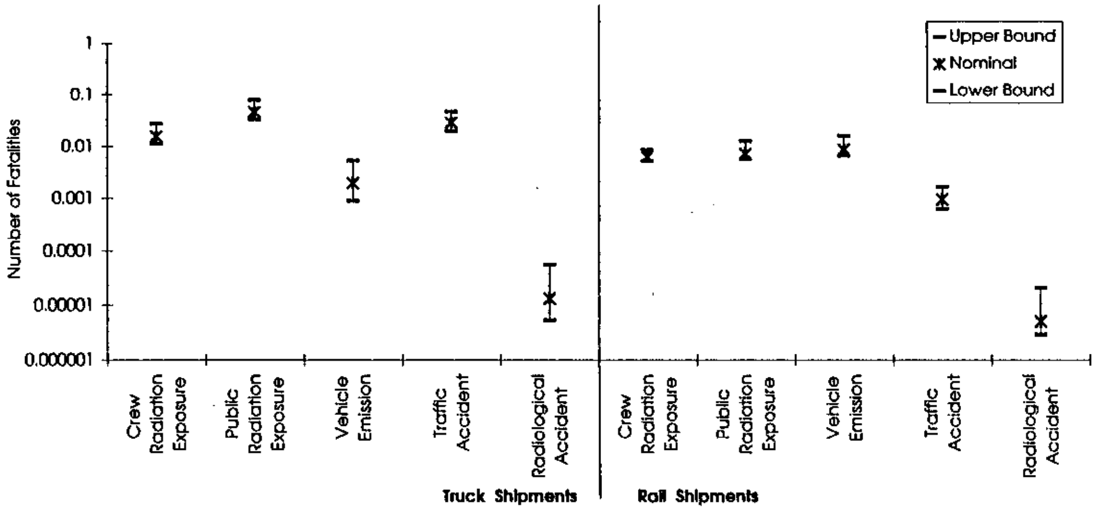
Management Site Impacts

As discussed in Chapter 2 of this EIS, Implementation Subalternative 2a reduces the quantity of foreign research reactor spent nuclear fuel to be managed to approximately 18,800 elements (compared to approximately 22,700 in the basic implementation of Management Alternative 1), but increases the rate of receipt to about 2,350 elements per year for an 8-year receipt period. This rate could challenge the capability of handling the incoming foreign research reactor spent nuclear fuel at a single site and could necessitate the use of both the Idaho National Engineering Laboratory and the Savannah River Site as near term foreign research reactor spent nuclear fuel management sites.

Incident-Free Impacts

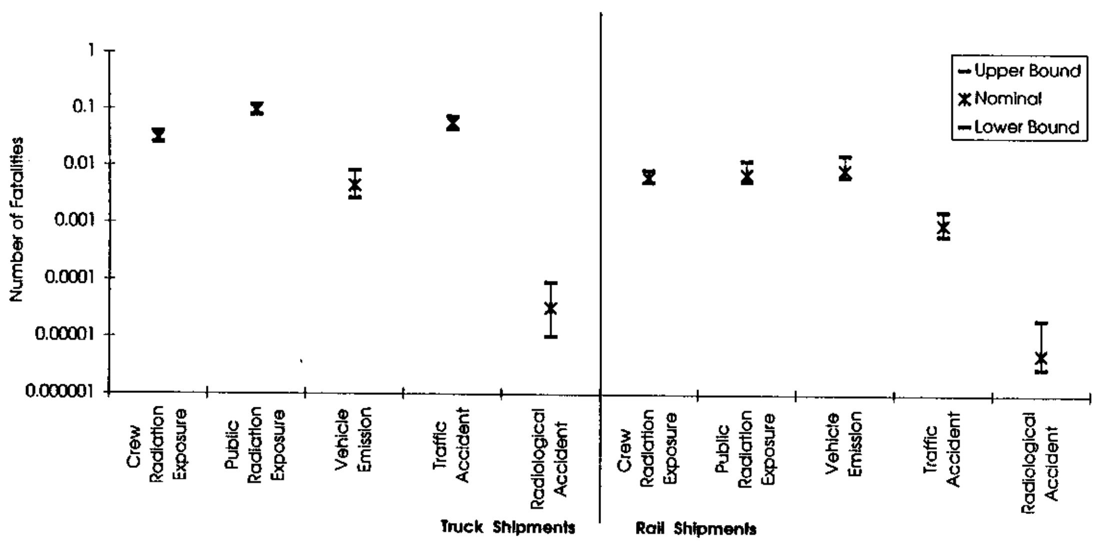
Based on the reduced number of foreign research reactor spent nuclear fuel elements that would be accepted under this subalternative, the worker population risk would be about 83 percent of that calculated for the basic implementation of Management Alternative 1. The maximally exposed worker risk was calculated in the same way as for the basic implementation of Management Alternative 1, with reduced handling time. If one worker received the maximum dose every year for eight years, his increased risk would be 0.016 LCF.

Some of the incident-free public risk depends on the amount of foreign research reactor spent nuclear fuel involved and some depends on the duration of each activity. The risk that accrues during receipt and handling can be scaled down by the factor of 83 percent from the basic implementation of Management Alternative 1, while the risk that accrues during storage is dependent only on the duration of the storage. The highest estimated incident-free public MEI risk in the basic implementation of Management Alternative 1 (1.4×10^{-7} LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (1.2×10^{-7} LCF).



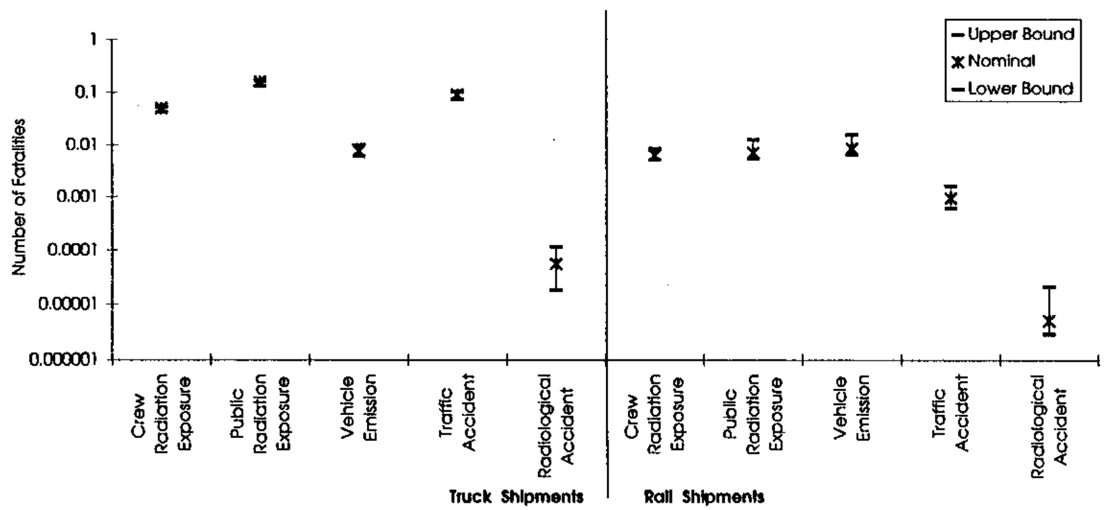
- *Same values apply to Programmatic SNF&INEL EIS 1992/1993 Planning Basis and INEL/SRS Regionalization alternatives*
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-14 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Decentralization Alternative



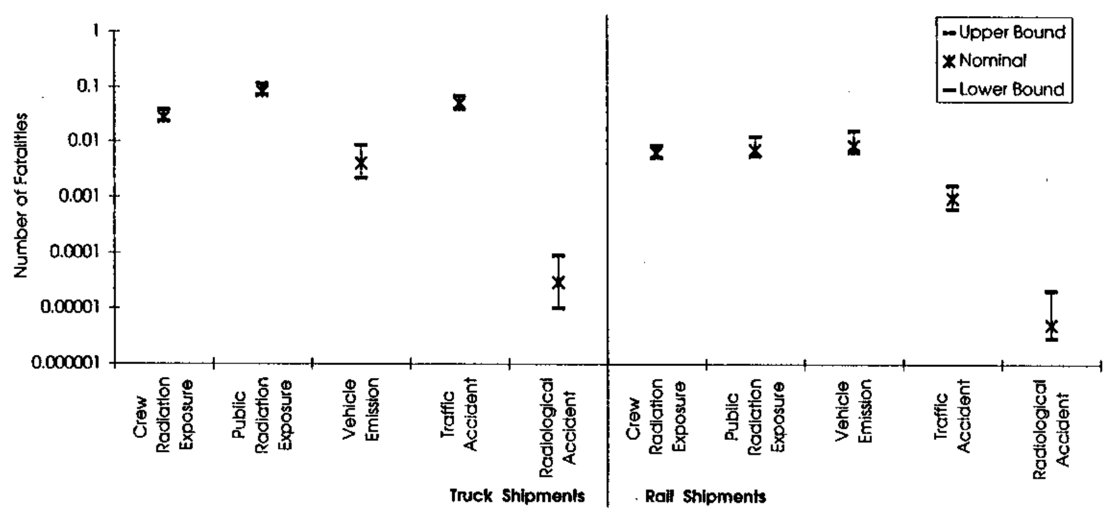
- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-15 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Regionalization by Fuel Type Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-16 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Savannah River Site Alternative



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-17 Range of Estimated Fatalities (Latent and Immediate) Under Implementation Subalternative 2a and the Programmatic SNF&INEL Final EIS Centralization to the Idaho National Engineering Laboratory Alternative

The highest estimated incident-free public population risk in Phase 1 of the basic implementation of Management Alternative 1 (0.00014 LCF) is due to 13 years of storage in L-Reactor Basin. The Phase 1 storage time in this subalternative would be slightly lower, and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.00013 LCF) is due to receipt and handling, so this component of the risk is reduced to 0.00011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.00025 LCF.

Impacts of Accidents Onsite

The highest estimated public MEI risk due to accident conditions in the basic implementation of Management Alternative 1 (0.000010 LCF) is due to receipt and handling, so it is reduced by the factor of 83 percent to yield the corresponding risk for this subalternative (0.0000083 LCF). This is higher than any other combination of Phase 1 or Phase 2 annual risk and duration.

The highest estimated population risk due to Phase 1 accidents in the basic implementation of Management Alternative 1 (0.096 LCF) is due to an accidental criticality in RBOF. This facility would be used for less time in this subalternative and the estimated risk could be reduced, but for simplicity and to be conservative, DOE and the Department of State did not reduce this component of the risk estimate compared to the basic implementation. The corresponding Phase 2 risk (0.013 LCF) is due to receipt and handling, so this component of the risk is reduced by the factor of 83 percent, down to 0.011 LCF for this subalternative. The sum of the Phase 1 and Phase 2 risks is 0.11 LCF.

Summary of the Impacts of Implementation Subalternative 2a

The principal impacts under this subalternative would be occupational and public health and safety impacts. These are presented in Table 4-38 in terms of the risk of death due to cancer during each of the four segments of this subalternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-38 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.016 LCF, which would apply to an onsite radiation worker. This individual would have a 1.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 1.2×10^{-7} LCF.

The highest estimated accident MEI risk is 0.0000083 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in one hundred thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

Table 4-38 Maximum Estimated Radiological Health Impacts of Implementation Subalternative 2a (Five-Year Policy)

	Risks (LCF)		
	Maximum Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00032	0	0.028
Accidents	4×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00032	0	0.0098
Accidents	1.6×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00032	0.20	0.064
Accidents	1.1×10^{-11}	0.00026	---
Site Activities			
Incident-Free	0.016	0.00025	0.17
Accidents	0.0000083	0.11	---
Highest Individual Risk			
Incident-Free	0.016	---	---
Accidents	0.0000083	---	---
Total Population Risk			
Incident-Free	---	0.20	0.27
Accidents	---	0.11	---

As shown in Table 4-38, the total incident-free population risk would be 0.20 LCF for the potentially exposed public, while the corresponding risk would be 0.27 LCF for workers. Thus, there would be an estimated 20 percent chance of incurring one additional LCF among the exposed general public, and a 27 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-38. There is about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.2.2 Implementation Subalternative 2b: Indefinite HEU/10-Year LEU Policy

Policy Considerations

The only difference between Implementation Subalternative 2b and the basic implementation of Management Alternative 1 would be to allow the acceptance of HEU spent nuclear fuel indefinitely from reactors with long-term lifetime cores, or from reactors whose operators for some reason (e.g., political) refuse to send their HEU spent nuclear fuel to the United States at this time. The exclusion of foreign research reactors that could be converted, but are not converted would be the same as in the basic implementation of Management Alternative 1. The amount of HEU spent nuclear fuel involved would also be the same as in the basic implementation of Management Alternative 1—only the timing would be different. The amount of HEU spent nuclear fuel that would be accepted after the policy period cannot be quantified because DOE and the Department of State do not know with certainty which countries would refuse to send their foreign research reactor spent nuclear fuel to the United States during the policy period of the basic implementation of Management Alternative 1. Nevertheless, this subalternative would provide a mechanism whereby DOE and the Department of State could increase the amount of U.S. origin HEU that could be recovered.

Impacts

The environmental impacts would be the same as, or slightly less than, those of the basic implementation of Management Alternative 1. Delaying the acceptance of a small fraction of the total amount of foreign research reactor spent nuclear fuel accepted would have a miniscule effect on the results presented in Section 4.2.

4.3.3 Implementation Alternative 3: Alternative Financing Arrangements

Under the basic implementation of Management Alternative 1, DOE and the Department of State would subsidize developing nations and charge developed nations a competitive rate. As discussed in Chapter 2, DOE and the Department of State have identified three potential financial arrangements:

- Subsidize all nations,
- Charge all nations the full cost of managing their spent nuclear fuel, and
- Subsidize developing nations and charge developed nations the full cost of managing their spent nuclear fuel.

Policy Considerations

Subsidizing all countries would be the most expensive for the United States. All the costs of transport, handling, storage, preparation for disposal, and disposal would be borne by the United States. The amount of HEU that would be accepted under this arrangement would likely be the same as under the basic implementation of Management Alternative 1.

Charging all countries the full cost of foreign research reactor spent nuclear fuel management would be the least expensive for the United States. All the costs would be borne by the foreign countries. Many developing countries probably would be unable to pay these high costs and this could lead to large quantities of HEU foreign research reactor spent nuclear fuel remaining in the countries least able to protect it. This could also lead to charges, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty. Even some developed countries might refuse to pay a full cost recovery fee, thus broadening the scope of problems this arrangement could cause.

Subsidizing developing countries and charging developed countries full cost of spent nuclear fuel management would be somewhat less expensive for the United States than the basic implementation of Management Alternative 1. Developing countries would be treated the same as in the basic implementation of Management Alternative 1, but developed countries would be charged more than in the basic implementation of Management Alternative 1. It is not clear how much more because the amount of a full cost recovery fee cannot be determined accurately at this time. Nevertheless, this increase over the internationally competitive rate could lead those nations which can reprocess to do so and perhaps to switch back to HEU fuel. Those nations in which reprocessing is not a viable option might force their reactors to shut down, and then charge, rightly or wrongly, that the United States was not complying with its obligations under Article IV of the Non-Proliferation Treaty.

Impacts

The different financial arrangements under this implementation alternative would have no direct effect on the environmental impacts of accepting and managing foreign research reactor spent nuclear fuel. Indirect effects are possible because, if the price is too high, some reactor operators may choose not to ship their spent nuclear fuel to the United States. This would reduce the amount of spent nuclear fuel accepted and thereby reduce the environmental impacts. It would be speculative, at best, to estimate the amount of spent nuclear fuel that might be excluded under this implementation alternative compared to the basic implementation of Management Alternative 1, so the changes in the environmental impacts cannot be quantified. It is clear however, that these changes would reduce overall environmental impacts in the United States during the policy period.

4.3.4 Implementation Alternative 4: Alternative Locations for Taking Title

Policy Considerations

The Price-Anderson Act applies to the shipments, independent of who holds title to the spent nuclear fuel. Thus, there is no change in the liability protection provided to the citizens of the United States, no matter where DOE takes title to the foreign research reactor spent nuclear fuel. Hence, there would be no change in the physical mode of shipping nor in the cost of shipping. Nevertheless, DOE and the Department of State are considering the following arrangements regarding the location for taking title to the foreign research reactor spent nuclear fuel:

- Taking title prior to shipment [i.e., at the foreign research reactor(s)],
- Taking title at the port(s) of entry, and
- Taking title at the DOE management site(s).

If DOE were to take title to the foreign research reactor spent nuclear fuel at the foreign research reactors, the liability protection afforded the citizens of the United States would not change, and the shipping arrangements would still be the same. However, DOE would then be liable for any mishaps that might occur in the foreign nations, or on the high seas. Thus, the potential liability to the United States might exceed the liability under the basic implementation of Management Alternative 1.

Taking title at the port(s) of entry would leave title in the hands of the foreign research reactor operators for the distance from the U.S. territorial waters limit to the port, thus potentially causing public concern about who would be liable to respond to any accident that might occur during that portion of the shipment. Similarly, taking title at the DOE management site would leave title in the hands of the foreign research reactor operators for an even greater distance within the United States, leading to even greater public concerns. These potential concerns would be borne of a misunderstanding because ownership does not affect shipping arrangements and precautions or liability protection. Nevertheless, it is likely that such concerns would exist.

Impacts

The environmental impacts (if any) of spent nuclear fuel shipments are not affected by the identity of the owner of the spent nuclear fuel. Therefore, the point of transfer of title is not a factor in determining environmental impacts.

4.3.5 Implementation Alternative 5: Wet Storage Technology for New Construction

Wet storage technology for new construction was considered instead of the dry storage technology contained in the basic implementation of Management Alternative 1, for all five potential foreign research reactor spent nuclear fuel management sites. The impacts during marine transport, port activities, and ground transport would be the same as in the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the analysis examined environmental topics including land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air quality, water quality, ecology, occupational health and safety, noise, utilities and energy, and waste management.

The means by which this alternative would be implemented at each management site are presented in Sections 2.6.5.3.1 through 2.6.5.3.5. The environmental impact analysis assumes that a new wet storage facility, which is described in Section 2.6.5.1.2, would be constructed at the sites to receive and store foreign research reactor spent nuclear fuel after the Phase 1 period. At the Savannah River Site, the alternative could also be implemented at the Barnwell Nuclear Fuels Plant (BNFP) and at the Hanford Site by the addition of facilities to the WNP-4 Spray Pond. These facilities are described in Appendix F, Section F.3. The analysis parallels in all respects the impact analysis performed for the new dry storage facility of the basic implementation of Management Alternative 1. It is presented in detail in Appendix F, Section F.4, with methodology and assumptions for radiological impacts given in Sections F.5 and F.6.

As in the basic implementation of Management Alternative 1, the analysis showed that this implementation alternative would not cause any major environmental impacts. Further, none of the environmental topics would clearly differentiate among the potential foreign research reactor spent nuclear fuel management sites.

4.3.5.1 Occupational and Public Health and Safety

As in the basic implementation of Management Alternative 1 (see Section 4.2.4.1) radiological exposures are presented as emissions-related impacts, handling-related impacts, and accident-related impacts.

Impacts to the Public of Incident-Free Management Site Activities

Table 4-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at each Phase 2 site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

The highest estimated Phase 1 public MEI and population risks for this alternative are identical to those for the basic implementation of Management Alternative 1. All possible Phase 1 MEI risks are lower than the highest estimated Phase 2 MEI risk in the next paragraph, so they will drop out. The highest Phase 1 component of the population risk is 0.00014 LCF in the basic implementation.

Among all the potential Phase 2 foreign research reactor spent nuclear fuel management sites, the maximum annual dose to the public from emissions is 0.06 mrem per year and 0.06 person-rem per year at the Oak Ridge Reservation for the MEI dose and the population dose, respectively. If it is assumed that receipt of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation could take place over a period of 3 years, the total MEI dose would be 0.18 mrem and the total population dose would be 0.18 person-rem. If it is further assumed that storage will continue for 30 years after the beginning of the receipt period, the total MEI dose from storage would be 1.4×10^{-5} mrem and the total population dose from storage would be 1.5×10^{-5} person-rem. The risks due to receipt and unloading would be much

Table 4-39 Annual Public Impacts for Receipt and Management of Foreign Research Reactor Spent Nuclear Fuel Under Implementation Alternative 5 (Wet Storage)

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Savannah River Site				
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.0000060
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
• New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at:				
• New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}
Nevada Test Site				
Receipt/Unloading at:				
• New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at:				
• New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

higher than those due to storage, so the maximum risk would be 0.18 mrem for the MEI and the sum of population doses would be 0.18 person-rem. The associated probabilities for incurring one LCF would be 9×10^{-8} LCF for the Phase 2 MEI risk and 0.00009 LCF for the Phase 2 population risk.

The maximum of the Phase 1 and Phase 2 incident-free public MEI risks is 9×10^{-8} LCF for this alternative. The sum of the Phase 1 and Phase 2 incident-free public population risks is 0.00023 LCF.

Impacts to Workers of Incident-Free Management Site Activities

As in the basic implementation of Management Alternative 1, workers would receive radiation doses during handling operations, such as receiving and unloading foreign research reactor spent nuclear fuel transportation casks at the site or transferring foreign research reactor spent nuclear fuel from one facility to another within the site. The methodology and assumptions for the analysis of this implementation alternative parallel that for the basic implementation of Management Alternative 1 as presented in Section 4.2.4.1 and Appendix F, Section F.5.

Table 4-40 presents the collective doses and risks that would be received by the members of the working crew, if that crew handled the total number of casks at the site.

Table 4-40 Handling-Related Impacts to Workers at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
Savannah River Site		
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phases 1 and 2: BNFP[a]	310	0.12
Idaho National Engineering Laboratory		
Phase 1: IFSF/CPP-749	257	0.10
Phases 1 and 2: New Wet Storage Facility	367	0.15
Phase 1: FAST	250	0.10
Phases 1 and 2: New Wet Storage Facility	360	0.14
Hanford Site		
Phase 2: New Wet Storage Facility or WNP-4 Spray Pond	109	0.04
Oak Ridge Reservation		
Phase 2: New Wet Storage Facility	109	0.04
Nevada Test Site		
Phase 2: New Wet Storage Facility	109	0.04

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

As seen from Table 4-40, the maximum total collective dose to workers handling foreign research reactor spent nuclear fuel at a single site would be 367 person-rem for the case analyzed at the Idaho National Engineering Laboratory, which assumes that all foreign research reactor spent nuclear fuel is in dry storage during Phase 1 and is transferred to a new wet storage facility for Phase 2. The associated probability for one LCF among the working crew would be 0.15. The highest dose to working crews for both phases in more than one site is 366 person-rem: 109 person-rem at one of the three Phase 2 sites plus 257 person-rem at the Idaho National Engineering Laboratory as the Phase 1 site. The associated probability for developing one LCF among the working crews of the two sites is 0.15.

Accident-Related Impacts

The accident scenarios analyzed for this implementation alternative are the same as those analyzed for the basic implementation of Management Alternative 1.

Table 4-41 presents the frequency and consequences of the accidents analyzed for each management site for this implementation alternative. Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at each candidate management site. Table 4-42 presents the annual risk estimates for wet storage.

The highest MEI or NPAI risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1 (2.6×10^{-6} LCF). The highest annual MEI or NPAI risk for Phase 2 would be 0.000005 LCF per year, which is the annual risk to the NPAI from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this MEI/NPAI risk would

Table 4-41 Frequency and Consequences of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

Site	Frequency (per yr)	Consequences							
		MEI		NPAI		Population		Worker	
		(mrem)	(LCF)	(mrem)	(LCF)	(person-rem)	(LCF)	(mrem)	(LCF)
Savannah River Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	3.5×10^{-9}	0.00039	2×10^{-10}	0.23	0.00012	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	17	0.0000085	9.5	0.0000048	370	0.19	1,600	0.00064
• Aircraft Crash	1×10^{-6}	4.1	0.0000021	0.98	4.9×10^{-7}	150	0.075	400	0.00016
BNFP									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	9×10^{-9}	0.00099	5×10^{-10}	0.028	0.000014	0.00080	3.2×10^{-10}
• Accidental Criticality[a]	0.0031	80	0.000040	75	0.000038	44	0.022	75	0.000030
• Aircraft Crash	1×10^{-6}	92	0.000046	31	0.000016	23	0.012	70	0.000028
Idaho National Engineering Laboratory									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	8×10^{-10}	0.0036	1.8×10^{-9}	0.43	0.00022	0.14	5.6×10^{-8}
• Accidental Criticality	0.0031	28	0.000014	30	0.000015	140	0.070	1,800	0.00072
• Aircraft Crash	1×10^{-6}	22	0.000011	9.8	0.0000049	250	0.13	400	0.00016
Hanford Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	6.5×10^{-8}	0.0033	1.7×10^{-9}	1.6	0.00080	0.25	1.0×10^{-7}
• Accidental Criticality	0.0031	64	0.000032	14	0.000007	740	0.37	3,600	0.0014
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
WNP-4 Spray Pond									
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.15	7.5×10^{-8}	0.0033	1.7×10^{-9}	1.3	0.00065	0.00024	9.6×10^{-11}
• Accidental Criticality[a]	0.0031	97	0.000049	76	0.000038	620	0.31	120	0.000048
• Aircraft Crash[b]	NA	NA	NA	NA	NA	NA	NA	NA	NA
Oak Ridge Reservation									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	3.6×10^{-7}	0.20	1.0×10^{-7}	16	0.0080	0.68	2.7×10^{-7}
• Accidental Criticality	0.0031	1,500	0.00075	3,300	0.0017	1,400	0.70	6,800	0.0027
• Aircraft Crash	1×10^{-6}	380	0.00019	600	0.00030	2,900	1.5	1,900	0.00076
Nevada Test Site									
New Wet Storage Facility									
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	2.7×10^{-8}	0.0016	8×10^{-10}	0.33	0.00017	0.10	4.0×10^{-8}
• Accidental Criticality	0.0031	88	0.000044	15	0.0000075	54	0.027	1,300	0.00052
• Aircraft Crash	1×10^{-6}	29	0.000015	4.2	0.0000021	61	0.031	290	0.00012

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

Table 4-42 Annual Risks of Accidents at Each Management Site Under Implementation Alternative 5 (Wet Storage)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Savannah River Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-7}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}
Idaho National Engineering Laboratory				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}
Hanford Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[b]	NA	NA	NA	NA
WNP-4 Spray Pond				
• Spent Nuclear Fuel Assembly Breach[a]	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality[a]	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[b]	NA	NA	NA	NA
Oak Ridge Reservation				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}
Nevada Test Site				
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

[a] *Emissions would be released through a tall stack, so workers would receive low doses.*

[b] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

NA = Not applicable

be 0.00015 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Taking the maximum of the Phase 1 and Phase 2 MEI risks yields 0.00015 LCF for the maximum MEI risk due to accidents.

The highest population risk for Phase 1 would be the same as under the basic implementation of Management Alternative 1, 0.096 LCF. The highest annual population risk for Phase 2 would be 0.0022 LCF per year, which is the annual risk to the public from an accidental criticality at the Oak Ridge Reservation. Assuming that foreign research reactor spent nuclear fuel could be managed at the

Oak Ridge Reservation for as long as 30 years, the Phase 2 component of this population risk would be 0.066 LCF. This is higher than any other combination of Phase 2 annual accident risks and associated durations in this implementation alternative. Adding the Phase 1 and Phase 2 population risks yields 0.16 LCF for the total population risk due to accidents.

4.3.5.2 Topics Not Discussed in Detail

Nonradiological impacts associated with the wet storage implementation alternative are similar to those for dry storage considered in the basic implementation of Management Alternative 1. They are discussed in detail in Appendix F, Section F.4.

Impacts at each management site typically associated with construction activities such as land use, socioeconomics, cultural resources, aesthetic and scenic resources, air quality, ecology, and noise are similar because: (1) both dry and wet storage facilities could be constructed at the same locations at each site; and (2) both facilities are approximately the same size. As indicated in Section 2.6.5.1, the construction of the wet storage facility would disturb approximately 2.8 ha (7 acres) of land while the construction of the dry storage facility would disturb 3.6 to 4.5 ha (9 to 11 acres). Specifically for the Savannah River Site, if the wet storage alternative is implemented using the BNFP facility there would be no impacts associated with construction activities.

Impacts at each management site typically associated with the operation of the facilities such as air quality, water quality, socioeconomics, utilities, and waste generation are also very similar as indicated in Section 2.6.5.1. The only notable difference is indicated in water use. The wet storage facility would use 1.5 million liters (409,000 gal) per year during the storage mode of the operation (over 30 years) compared to 0.9 million liters (238,000 gal) per year used by the dry storage facility over the same period. This difference, however, is small compared to typical water consumption rates at the sites: 1.14 billion liters (300 million gal) per year at the Nevada Test Site to 88 billion liters (23.2 billion gal) per year at the Savannah River Site.

4.3.5.3 Summary of the Impacts of Implementation Alternative 5

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-43 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-43 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

Table 4-43 Maximum Estimated Radiological Health Impacts of Implementation Alternative 5 (Wet Storage)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.22	0.071
Accidents	1.4×10^{-11}	0.00028	---
Site Activities			
Incident-Free	0.026	0.00023	0.15
Accidents	0.00015	0.16	---
Highest Individual Risk			
Incident-Free	0.026	---	---
Accidents	0.00015	---	---
Total Population Risk			
Incident-Free	---	0.22	0.27
Accidents	---	0.16	---

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 9×10^{-8} LCF.

The highest estimated accident MEI risk is 0.00015 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than two in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-43, the total incident-free population risk would be 0.22 LCF for the potentially exposed public, while the corresponding risk would be 0.27 LCF for workers. Thus, there would be an estimated 22 percent chance of incurring one additional LCF among the exposed general public, and a 27 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-43. There is about a 14 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

4.3.6 Implementation Alternative 6: Near Term Chemical Separation in the United States

As discussed in Section 2.2.2.6, this implementation alternative involves conventional chemical separation in existing facilities at either the Savannah River Site or the Idaho National Engineering Laboratory. The facilities at the Savannah River Site are limited to chemically separating the aluminum-based foreign research reactor spent nuclear fuel. After some upgrading, the facilities at the Idaho National Engineering Laboratory would have the capability to chemically separate all the foreign research reactor spent nuclear fuel.

4.3.6.1 Implications of Chemical Separation for U.S. Nonproliferation Policy

As a matter of policy, the United States does not currently engage in reprocessing or chemical separation to extract plutonium for civilian or military purposes. U.S. policy is also not to encourage the civilian use of plutonium and to explore means to limit the stockpiling of plutonium from civil nuclear programs. This alternative nonetheless considers scenarios whereby the United States might engage in future chemical separation of foreign research reactor spent nuclear fuel. If a decision were made pursuant to this EIS to chemically separate some or all of the foreign research reactor spent nuclear fuel, the limited amount of plutonium in the spent fuel would not be separated. Rather it would be left in, and disposed of with, the high-level radioactive wastes produced during the chemical separation operation.

Two alternatives are evaluated for handling the highly enriched uranium in the spent fuel, either to blend it down to low enriched uranium (the preferred alternative, if any chemical separation is undertaken), or to separate it as HEU and place it in safe, secure storage. Chemical separation of foreign research reactor spent nuclear fuel, with blending down of the separated uranium, would, in fact, result in a reduction in the amount of HEU – a major goal of the U.S. Nuclear Weapons Nonproliferation Policy announced in September 1993. Despite this fact, there is a concern that other states may perceive only that the U.S. has restarted reprocessing.

For example, the potential exists that other states (e.g., Iran), might use the restart of reprocessing in the United States as an excuse to continue current programs or begin new ones – activities that would run counter to U.S. nuclear weapons nonproliferation interests. The implications in North Korea, where the United States has been actively working to create a nonreprocessing zone, as well as in other states, could complicate current U.S. nonproliferation activities.

4.3.6.2 General Assumptions and Analytic Approach

Potential impacts at the Savannah River Site and the Idaho National Engineering Laboratory were estimated separately. The impacts due to chemical separation and associated onsite activities would be in addition to those due to marine transport, port activities, and ground transport.

As discussed in Section 2.2.2.6, DOE and the Department of State have analyzed four possible chemical separation subalternatives under this implementation alternative. These four subalternatives, with spent nuclear fuel amounts and estimated facility run durations are:

	Amount (MTHM)	***Duration (Years)***
Savannah River Site (only aluminum-based spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	18.2	13
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	51	13
Idaho National Engineering Laboratory (aluminum-based and TRIGA spent nuclear fuel)		
• Foreign research reactor spent nuclear fuel only	19.2	12
• Foreign research reactor spent nuclear fuel plus other spent nuclear fuel	65	12

The duration of chemical separation operations dedicated to foreign research reactor spent nuclear fuel is driven by the rate of foreign research reactor spent nuclear fuel receipt at the Savannah River Site or the Idaho National Engineering Laboratory. The facility run durations at Savannah River Site are both up to 13 years, whether the facilities would be chemically separating only the 18.2 MTHM of foreign research reactor spent nuclear fuel or the 51 MTHM of spent nuclear fuel. Because the additional spent nuclear fuel would be chemically separated at the same time as the foreign research reactor spent nuclear fuel in a parallel process, only the combined impacts will be used to determine the risks associated with the overall operations. There are other nuclear materials, such as the Mark-31 targets currently stored at the Savannah River Site, which could also be chemically separated. These nuclear materials are not included in this implementation alternative, but they are covered under cumulative impacts. The impacts of running the facilities are based on conservative assumptions regarding incident-free annual emissions and possible accident releases which cover this range of throughputs.

The facility run durations at the Idaho National Engineering Laboratory are estimated to be up to 12 years. Furthermore, the same type of conservative assumptions regarding incident-free emissions and accidental releases are applied to calculate the environmental impacts.

As discussed in Section 2.2.2.6, the implementation component of uranium disposition has policy implications. The separated LEU could be returned to the commercial sector for reuse as reactor fuel. The HEU could be blended down to LEU or it could be processed directly to an oxide and stored. If a decision is made to chemically separate this spent nuclear fuel, it would be DOE's preference to blend down the HEU to LEU and thus preclude the possibility of this material ever being used in a nuclear weapon.

4.3.6.3 Marine Transport Impacts

The marine transport impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.1.

4.3.6.4 Port Activities Impacts

The port activities impacts of this implementation alternative would be identical to those of the basic implementation of Management Alternative 1, as discussed in Section 4.2.2.

4.3.6.5 Ground Transport Impacts

The impacts due to ground transport of foreign research reactor spent nuclear fuel in this implementation alternative would be slightly lower than those of the basic implementation of Management Alternative 1, because the Phase 2 intersite shipments would not occur.

If the aluminum-based foreign research reactor spent nuclear fuel were chemically separated at the Savannah River Site it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period. The TRIGA foreign research reactor spent nuclear fuel would be transported to either of the two sites for management and it would not be transported again for the duration of the 40-year program period.

Similarly, if all the foreign research reactor spent nuclear fuel were chemically separated at the Idaho National Engineering Laboratory, it could not then be transported to another management site as spent nuclear fuel. The high-level waste resulting from this chemical separation would be managed onsite for the duration of the 40-year program period.

Impacts of Incident-Free Ground Transport

The impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates was due to two factors: the option of using truck or rail to transport spent nuclear fuel and combinations of management sites that created varying cask shipment numbers and distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCF for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCF from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and the potential foreign research reactor spent nuclear fuel management sites that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The consequences of the maximum foreseeable offsite transportation accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced amount of ground transport, so the MEI risk is reduced to 1.3×10^{-11} LCF.

4.3.6.6 Impacts at the Potential Foreign Research Reactor Spent Nuclear Fuel Management Sites

DOE and the Department of State evaluated near term chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory for five key types of impacts: (1) Socioeconomics, (2) Air Quality, (3) Water Quality, (4) Occupational and Public Health and Safety, and (5) Waste Management. The other impacts are all the same as those described in the basic implementation of Management Alternative 1. The analytic approach was to use the results published in the Programmatic SNF&INEL Final EIS (DOE, 1995c) and the Interim Management of Nuclear Materials Final EIS (DOE, 1995a) whenever possible. Usually, these results can be adopted directly.

4.3.6.6.1 Socioeconomics

Savannah River Site

The chemical separation facilities at the Savannah River Site were last operated in 1992. The facilities are in a warm standby condition and are currently fully staffed. Use of these facilities would not have a notable net impact upon employment or the regional economy.

Idaho National Engineering Laboratory

The chemical separation facilities at the Idaho National Engineering Laboratory were last operated in 1990 and are currently in the process of being cleaned out in preparation for decommissioning. Some staff would need to be added eventually, but the use of these facilities would not have a notable net impact upon employment or the regional economy.

4.3.6.6.2 Air Quality

Savannah River Site

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Savannah River Site in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions would be small increases over baseline site-wide totals and within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has analyzed the expected airborne radiological emissions from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These radiological emissions are presented in Table 4-44 (Grainger, 1995). The health effects from these airborne emissions are discussed in Section 4.3.6.6.4 below.

Table 4-44 Annual Incident-Free Airborne Radiological Emissions at the Savannah River Site that Contribute to the Offsite Dose[a]

Element	*Ci/yr*
Tritium	57.8
Cesium-134	0.002
Cesium-137	0.12
Curium-242/244	0.12
Cerium-144	0.0059
Americium-241	0.016
Cobalt-60	0.000000053
Plutonium-238	0.078
Plutonium-239	0.020
Strontium-89/90	0.17
Iodine-131	0.0053
Uranium-235/238	0.039
Antimony-125	0.018
Ruthenium-106	0.20

[a] *Krypton-85 would be released at an estimated rate of 120,000 Ci/yr*

Source: Grainger, 1995

Krypton-85 emissions are not included in Table 4-44 because these releases are not normally measured or calculated. The health effects resulting from krypton-85 releases are very low compared to those resulting from other isotopes that are being measured. Krypton is an inert gas with no affinity for biological systems, so it does not adhere to the lungs if inhaled. The radioactive isotope of krypton would cause such a low level of harm to the population near the Savannah River Site because it remains in the human body for only very brief periods of time. The total amount of krypton-85 that would be contained in all of the aluminum-based foreign research reactor spent nuclear fuel is conservatively estimated to be 1.5 x 10^6 curies. Assuming this is released gradually over the 12-year reprocessing period, the annual emission rate would be 1.2 x 10^5 curies per year.

Idaho National Engineering Laboratory

Incident-Free Nonradiological Emissions

DOE has analyzed the expected nonradiological emissions from its chemical separations facilities at the Idaho National Engineering Laboratory in the Programmatic SNF&INEL Final EIS (DOE, 1995c). All estimated emissions are within regulatory limits (DOE, 1995c).

Incident-Free Radiological Emissions

DOE has also analyzed the expected radiological emissions from the Idaho National Engineering Laboratory chemical separations facilities in the Programmatic SNF&INEL Final EIS (DOE, 1995c). These are presented in Table 4-45. The radiological emission rates were estimated using conservative engineering calculations based on knowledge of the proposed activity. These emission rates are representative of emissions that could occur during Implementation Alternative 6 at the Idaho National Engineering Laboratory. Human health consequences are discussed in Section 4.3.6.6.4.

Table 4-45 Annual Incident-Free Airborne Radiological Emissions at the Idaho National Engineering Laboratory

Element	*Ci/yr*
Tritium + Carbon-14	3,100
Cesium-134 + Cesium-137	0.18
Cobalt-60	0.0000019
Plutonium	0.0077
Strontium-90 + Yttrium-90	0.058
Krypton-85	500,000
Antimony-125	16
Iodine-129 + Iodine-131	0.44
Others	0.21

Source: DOE, 1995b

4.3.6.6.3 Water Quality

Savannah River Site

DOE has analyzed the expected liquid radiological releases from the Savannah River Site chemical separations facilities in the Interim Management of Nuclear Materials EIS (DOE, 1995a). These releases are presented in Table 4-46 (Grainger, 1995). The health effects from these liquid releases are discussed in Section 4.3.6.6.4 below.

Idaho National Engineering Laboratory

Chemical separation activities at the Idaho National Engineering Laboratory would not affect water quality because the facility designs would prevent any accidental or incident-free discharge of liquid effluents (DOE, 1995c).

Table 4-46 Annual Incident-Free Liquid Radiological Releases at the Savannah River Site

Element	Ci/yr
Tritium	1.29
Strontium-89/90	0.013
Ruthenium-103/106	0.012
Cesium-137	0.033
Promethium-147	0.045

4.3.6.6.4 Occupational and Public Health and Safety

Potential exposures to workers and the public due to chemical separation activities were analyzed at both the Savannah River Site and the Idaho National Engineering Laboratory (DOE, 1995c). To estimate health effects, this analysis defined three receptor groups:

- onsite workers assigned to operations involving spent nuclear fuel,
- 1994 offsite population residing within an 80-km (50-mi) radius of the chemical separation facilities (exposure via air), and
- offsite population whom management site surface-water emissions could affect.

Each of these three receptor groups would receive an annual maximum individual dose and an annual population dose. The maximally exposed worker dose would be limited by regulation to 5,000 mrem per year, as in the basic implementation of Management Alternative 1.

Savannah River Site

Incident-Free Impacts at the Savannah River Site

The highest estimated incident-free dose rates for conventional chemical separation operations at the Savannah River Site are presented in Table 4-47 (DOE, 1995a). These chemical separation operations could include activities related to blending the separated HEU down to LEU and converting all LEU into an oxide suitable for long-term storage. Values in Table 4-47 represent the estimated dose rates due to these activities, including actual chemical separation, blending down, and conversion to oxide. Multiplying these values by the estimated program duration of 13 years yields the doses presented in Table 4-48. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-48. If the HEU were not blended down, but rather converted directly to oxide, the worker population dose would be higher because the conversion to oxide would take place in the Uranium Solidification Facility. In this facility, the workers would be closer to the uranium.

Table 4-47 Incident-Free Radiation Dose Rates Due to Chemical Separation at the Savannah River Site

	Maximum Individual Dose Rate (mrem/yr)	*Population Dose Rate (person-rem/yr)*
Public		
Via Air	0.66	27
Via Water	0.0098	0.033
Workers	5,000[a]	21

[a] *Assumed to be equal to the regulatory limit*

Table 4-48 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Savannah River Site

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	8.6	0.0000043	351	0.18
Via Water	0.13	6.4×10^{-8}	0.43	0.00021
Workers	65,000	0.026	273	0.11

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-8. The risks of storage at RBOF are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at RBOF for the full 13 years, the public MEI and population risks would be 7.1×10^{-10} LCF and 0.000036 LCF, respectively. These risks are much lower than the corresponding values in Table 4-48.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is equal to the 0.026 LCF in Table 4-48.

For the public, the estimated MEI risk from incident-free chemical separation activities would be 0.0000043 LCF. This risk means that an individual who lives at the Savannah River Site boundary would have an additional chance of less than one in one hundred thousand of incurring an LCF.

The handling-related worker population risk at RBOF is 0.10 LCF (see Table 4-14). This must be added to the 0.11 LCF from Table 4-48 to obtain the estimate of worker population risk due to chemical separation of foreign research reactor spent nuclear fuel. The estimated population risk for workers, including the handling-related risk, is 0.21 LCF, so there would be an approximately 21 percent chance of one additional LCF among the radiation workers.

The estimated total public population risk from chemical separation activities would be 0.18 LCF (see Table 4-48), which means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the Savannah River Site due to incident-free chemical separation activities.

Impacts of Chemical Separations Accidents at the Savannah River Site

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Savannah River Site (DOE, 1995a), including a hydrogen explosion in a high-level waste tank, an unpropagated fire in a solution vessel, two kinds of inadvertent transfers of solutions, a coil and tube failure in the cooling system, a nuclear criticality, a "red oil" explosion, a severe earthquake, and a tornado. The annual risks for the accident with the highest estimated combination of frequency and consequence are presented in Table 4-49. The most severe accident scenario is an unpropagated fire in a solution vessel. Multiplying these results by the estimated program duration of 13 years yields the risks presented in Table 4-50.

Table 4-49 Annual Impacts of Chemical Separation Accidents at the Savannah River Site

	Accident Frequency (per year)	Consequences (LCF)		Risks (LCF/yr)	
		Maximum Individual	Population	Maximum Individual	Population
Unpropagated Fire					
• Public	0.02	0.00018	1.3	0.0000036	0.026
• Workers	0.02	0.00086	---	0.000017	---

Table 4-50 Impacts of Accidents During Chemical Separation Operations at the Savannah River Site

	Maximum Individual Risk (LCF)	Population Risk (LCF)
Public	0.000047	0.34
Workers	0.00022	---

These results indicate that the estimated public MEI risk due to the chemical separation accidents is 0.000047 LCF. The estimated public population risk due to chemical separation accidents is 0.34 LCF. These risks must be combined with the risks of receiving/unloading and temporarily storing the foreign research reactor spent nuclear fuel, which were presented in Table 4-24. Assuming the foreign research reactor spent nuclear fuel would be received/unloaded and stored at RBOF for 13 years, the public MEI and population risks would be 0.0000026 LCF and 0.096 LCF, respectively.

The maximum of the two estimated accident-related MEI risks is 0.000047 LCF. This means that this hypothetical individual would have an additional chance of incurring an LCF of less than one in ten thousand.

The sum of the two population risks is 0.43 LCF. This means there would be an approximately 43 percent chance that one additional LCF would occur in the public population near the Savannah River Site due to accident conditions.

Idaho National Engineering Laboratory

Incident-Free Impacts at Idaho National Engineering Laboratory

The incident-free radiation dose rates for chemical separation at the Idaho National Engineering Laboratory are presented in Table 4-51 (DOE, 1995c). Multiplying these values by the estimated program duration of 12 years yields the doses presented in Table 4-52. These doses are converted into risks of LCF by applying the appropriate conversion factors and these results are also presented in Table 4-52.

Table 4-51 Incident-Free Radiation Dose Rates due to Chemical Separation at the Idaho National Engineering Laboratory

	Maximum Individual Dose Rate (mrem/yr)	Population Dose Rate (person-rem/yr)
Public		
Via Air	0.048	0.39
Via Water	0.0	0.0
Workers	5,000[a]	18

[a] *Assumed to be equal to the regulatory limit*

Table 4-52 Radiological Health Impacts Due to Incident-Free Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Dose (mrem)	*Maximum Individual Risk (LCF)*	*Population Dose (person-rem)*	*Population Risk (LCF)*
Public				
Via Air	0.58	2.9×10^{-7}	4.7	0.0024
Via Water	0.0	0.0	0.0	0.0
Workers	60,000	0.024	216	0.086

These risks must be combined with the risks of receiving/unloading the casks. Risks to the public were presented earlier in this chapter in Table 4-9. The risks of storage are also presented, but they are very low compared to those of receipt/unloading. Assuming the foreign research reactor spent nuclear fuel would be received at FAST for the full 13 years, the public MEI and population risks would be 2.5×10^{-9} LCF and 0.000021 LCF, respectively. These risks are much lower than the corresponding values in Table 4-52.

The handling-related risks to workers were presented earlier in this chapter. Under the conservative assumptions in the basic implementation of Management Alternative 1, the maximally exposed worker risk due to handling could be as high as 0.026 LCF which is higher than the 0.024 LCF in Table 4-52.

For the public, the estimated MEI risk due to incident-free chemical separation activities at the Idaho National Engineering Laboratory is less than one millionth of an LCF, which means that an individual who lives at the Idaho National Engineering Laboratory boundary would have an additional chance of less than one in a million of incurring an LCF.

The handling-related worker population risk at the Idaho National Engineering Laboratory is 0.10 LCF, from Table 4-15. This must be added to the 0.086 LCF from Table 4-52 to obtain the estimate of worker population risk due to incident-free chemical separation of foreign research reactor spent nuclear fuel, 0.19 LCF. This is near zero, so zero LCF would be expected among the radiation workers.

The estimated total public population risk at the Idaho National Engineering Laboratory is about 0.0024 LCF, which is much less than one LCF.

Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

DOE has analyzed the impacts of reasonably identifiable accidents due to chemical separation activities at the Idaho National Engineering Laboratory (DOE, 1995c). The accident with the highest estimated combination of frequency and consequence would be an inadvertent nuclear criticality during chemical separation. The accident frequency, consequences, and annual risks for this accident are presented in Table 4-53. Multiplying these results by the estimated program duration of 12 years yields the risks presented in Table 4-54.

Table 4-53 Annual Impacts of Chemical Separation Accidents at the Idaho National Engineering Laboratory

	Accident Frequency (per/yr)	*Consequences (LCF)*		*Risks (LCF/yr)*	
		Maximum Individual	*Population*	*Maximum Individual*	*Population*
Inadvertent Criticality					
Public	0.001	0.000025	0.0028	2.5×10^{-8}	0.0000028
Workers	0.001	0.0036	---	0.0000036	---

Table 4-54 Impacts of Accidents During Chemical Separation Operations at the Idaho National Engineering Laboratory

	Maximum Individual Risk (LCF)	*Population Risk (LCF)*
Public	3.0×10^{-7}	0.000034
Workers	0.000044	---

The highest estimated public MEI risk is 3.0×10^{-7} LCF, which means that an individual living at the management site boundary would have an additional chance of incurring an LCF of less than one in a million.

The highest estimated public population risk is 0.000034 LCF, which is much less than one LCF.

4.3.6.6.5 Waste Management

Savannah River Site

DOE has analyzed the wastes that would be generated from the aluminum-based foreign research reactor spent nuclear fuel and from an additional inventory of aluminum-based spent nuclear fuel during chemical separation activities. High-level waste, saltstone, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Savannah River Site. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. These aluminum-based spent nuclear fuel elements are similar to DOE's Mark 16/22 spent nuclear fuel elements at the Savannah River Site.

High-level liquid waste would be transferred to the F/H-Area Tank Farm for volume reduction and then to the Defense Waste Processing Facility for conversion into a borosilicate glass form suitable for prolonged storage. The high-level glass waste that would result from chemically separating the 18.2 MTHM (approximately 17,800 elements) of foreign research reactor spent nuclear fuel in this implementation subalternative would fill about 72 canisters (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 200 canisters. These canisters would be managed with the estimated 5,717 canisters that the Savannah River Site expects to produce from the existing onsite inventory of liquid high-level waste (WSRC, 1995). Each canister will contain approximately 40,000 curies of radioactivity (DOE, 1994a). The representative radionuclide composition of the waste glass is presented in Table 2.11 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be cesium-137, strontium-90, and their daughters. DOE expects that this waste form would be acceptable for disposal in a geologic repository.

Saltstone would be produced during the vitrification of high-level waste at the Defense Waste Processing Facility. An estimated 4,000 m^3 (140,000 ft^3) would be generated from the 18.2 MTHM of foreign research reactor spent nuclear fuel and would be disposed of onsite (Dupont, 1996). Scaling this result up to include the total inventory of 51 MTHM yields an estimate of about 11,300 m^3 (400,000 ft^3). This is much less than the maximum estimated cumulative saltstone to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 625,211 m^3 (about 22,000,000 ft^3) (DOE, 1994b). The saltstone would contain far less radioactivity than the high-level waste glass: approximately 0.1 curie per cubic meter (DOE, 1994a). The approximate composition of the saltstone in terms of specific radionuclides is presented in Table C.5 of the Integrated Data Base Report-1993 (DOE, 1994a). The radionuclides that contribute most of the radioactivity would be promethium-147 until about 2000, then strontium-90 and its daughter thereafter.

Transuranic waste would not be generated during the chemical separation activities of foreign research reactor spent nuclear fuel (DOE, 1995a). The trace amounts of transuranic elements would not be removed from the waste stream, so they would be included in the high-level waste. If the Taiwan Research Reactor spent nuclear fuel (included in the total inventory of 51 MTHM) is chemically separated and the transuranic elements removed, then an estimated 832 m^3 (about 29,400 ft^3) of transuranic waste would be generated (DOE, 1995a). This is much less than the maximum estimated cumulative transuranic waste to be generated at the Savannah River Site during the 10-year period from 1995 through 2004, which would be 9,426 m^3 (about 333,000 ft^3) (DOE, 1994b).

Hazardous/mixed waste would also be produced under this implementation subalternative. An estimated 104 m^3 (about 3,700 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a). This is much less than the maximum estimated cumulative mixed waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 14,720 m^3 (about 520,000 ft^3) (DOE, 1994b).

Solid low-level waste would also be produced under this implementation subalternative. An estimated 74,000 m^3 (about 2,600,000 ft^3) would be generated during 13 years of chemical separation operations (DOE, 1995a) and would be disposed of onsite. This is much less than the maximum estimated cumulative low-level waste to be generated throughout the entire Savannah River Site during the 10-year period from 1995 through 2004, which would be 397,177 m^3 (about 14,000,000 ft^3) (DOE, 1994b).

Idaho National Engineering Laboratory

DOE has also analyzed the wastes that would be generated from the foreign research reactor spent nuclear fuel and from an additional inventory of spent nuclear fuel during chemical separations activities at the Idaho National Engineering Laboratory. High-level waste, low-level grout, transuranic waste, hazardous/mixed waste, and low-level waste would be generated under this implementation subalternative. All these wastes would be managed along with similar wastes at the Idaho National Engineering Laboratory. The alternatives for managing all DOE wastes have been evaluated in the Draft Waste Management Programmatic EIS (DOE, 1995b).

The estimates of waste volumes that would be generated under this implementation subalternative are based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 56 canisters of high-level waste glass would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel yields 90 canisters. Scaling up further to the total inventory of 65 MTHM yields an estimate of about 300 canisters. These canisters would be managed along with the estimated 8,500 canisters Idaho National Engineering Laboratory expects to produce from the existing inventory of high-level waste onsite (DOE 1995b). Although the waste form has not been determined yet, each canister is estimated to contain approximately 22,000 curies of radioactivity (DOE, 1994a). The composition of the waste form in terms of specific radionuclides has not been determined yet, but it is reasonable to expect it to be similar to that of the glass at the Savannah River Site. DOE expects that the waste form would be acceptable for disposal in a geologic repository.

Another possibility exists if large quantities of nonaluminum-based spent nuclear fuels are being chemically separated in these facilities. Some aluminum is necessary to produce the stable waste form, and the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel could satisfy this requirement. In this case, the chemical separation of the aluminum-based spent nuclear fuel would not increase the number of canisters that would be generated at the Idaho National Engineering Laboratory.

The estimates of low-level waste grout that would be generated under this implementation subalternative are also based on comparisons with similar operations on similar spent nuclear fuels. For 12 MTHM of aluminum-based foreign research reactor spent nuclear fuel, 1,280 m^3 (about 45,000 ft^3) of low-level waste grout would be generated (Denney, 1995). Scaling up to the 19.2 MTHM of foreign research reactor spent nuclear fuel in this implementation subalternative yields about 2,000 m^3 (70,629 ft^3). Scaling up further to the total inventory of 65 MTHM yields an estimate of about 6,900 m^3 (about 245,000 ft^3). This grout would be managed along with the other grout the Idaho National Engineering Laboratory would produce onsite. The grout is expected to contain far less radioactivity than the high-level waste glass/ceramic: much less than one curie per cubic meter. The composition of the grout in terms of all the specific radionuclides has not been determined yet, but the major radioactive constituents would be cesium-137 and strontium-90. The cesium-137 and strontium-90 concentrations in the grout are expected to be about 0.034 and 0.0093 curies per cubic meter, respectively (Bendixsen, 1995).

Transuranic waste would not be generated during chemical separation of the foreign research reactor spent nuclear fuel. Furthermore, the Idaho National Engineering Laboratory would not separate the transuranic elements from the Taiwan Research Reactor spent nuclear fuel if it were transported there from the Savannah River Site. Therefore, no transuranic waste would be generated during chemical separation of the additional inventory of spent nuclear fuel. The estimated amount of cumulative transuranic waste for 10 years with minimum waste management at the Idaho National Engineering Laboratory is 67,000 m^3 (about 2,400,000 ft^3) (DOE, 1995c).

Hazardous/mixed waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 96 m^3 (3,400 ft^3) would be generated during 12 years of chemical separation operations. This is much less than the estimated 29,000 m^3 (1,020,000 ft^3) of cumulative hazardous and mixed waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c).

Solid low-level waste would also be produced under this implementation subalternative. Assuming a waste generation rate about equal to the rate at the Savannah River Site, an estimated 68,300 m^3 (2,400,000 ft^3) would be generated during 12 years of chemical separation operations and would be disposed of onsite. This is more than the estimated 47,000 m^3 (1,660,000 ft^3) of low-level waste to be generated throughout the entire Idaho National Engineering Laboratory during the next 10 years with minimum waste management (DOE, 1995c). The Idaho National Engineering Laboratory would treat the waste at the Waste Experimental Reduction Facility and send it to the Radioactive Waste Management Complex for onsite disposal.

4.3.6.7 Summary of the Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

The principal impacts under this implementation alternative would be occupational and public health and safety impacts. These are presented in Table 4-55 in terms of the risk of death due to cancer during each of the four segments of this implementation alternative. It also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. The marine transport, port activities, and ground transport impacts are identical to the basic implementation of Management Alternative 1. The management site activity impacts were derived by comparing, and summing as appropriate, the handling impacts of the basic implementation of Management Alternative 1 and the impacts of chemical separation dedicated to foreign research reactor spent nuclear fuel. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-55 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have approximately a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The population risks were calculated by summing the appropriate spent nuclear fuel handling risks from the basic implementation of Management Alternative 1 with the risks of chemical separation at each management site and selecting the largest value. For example, the incident-free worker population risk of 0.21 LCF is the largest sum of the risks from that estimated for spent nuclear fuel handling operations under Phase 1 of the basic implementation of Management Alternative 1 and the estimated risk due to chemical separation dedicated to foreign research reactor spent nuclear fuel at the Savannah River Site or the Idaho National Engineering Laboratory. The sum of the above risks at the Idaho National Engineering Laboratory is 0.19 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-15) and 0.086 LCF from chemical separation], and the corresponding value at the Savannah River Site is 0.21 LCF [0.10 LCF from Phase 1 of the basic implementation (Table 4-14) and 0.11 LCF from chemical separation].

As shown in Table 4-55, the total incident-free population risk would be 0.39 LCF for the potentially exposed public, while the corresponding risk would be 0.32 LCF for workers. Thus, there would be an estimated 39 percent chance of incurring 1 additional LCF among the exposed general public, and a 32 percent chance of incurring 1 additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-55. DOE and the Department of State estimate there could be about a 13 percent chance that a truck driver or member of the public could die in a traffic accident associated with this implementation alternative. This death would be unrelated to the radioactive nature of the cargo.

Table 4-55 Maximum Estimated Radiological Health Impacts of Implementation Alternative 6 (Near Term Conventional Chemical Separation)

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 0.000029	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2×10^{-10}	0.000029	---
Ground Transport			
Incident-Free	0.00052	0.21	0.065
Accidents	1.3×10^{-11}	0.00014	---
Site Activities			
Incident-Free	0.026	0.18	0.21
Accidents	0.000047	0.43	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.39	0.32
Accidents	----	0.43	----

4.3.7 Implementation Alternative 7: New Developmental Treatment and/or Packaging Technologies

The environmental impacts of the developmental treatment and/or packaging technologies cannot be estimated with confidence at this time because the technologies and procedures are still under development. Implementation of certain of these technologies would require new facilities and thus would generate all the impacts associated with construction. Appropriate NEPA documentation would be prepared to support a decision on implementation of a new technology. The developmental treatment and/or packaging technologies are described in Chapter 2, Section 2.2.2.7.

The date at which a new facility would be operational is highly uncertain. A fairly simple technology implemented in existing facilities could be operational by 2000. On the other hand, the technology development, NEPA analysis, facility construction, and startup could take about 15 years for a complex technology. Thus, DOE could choose to implement one of the accept-and-store alternatives, in parallel with this alternative to prepare the foreign research reactor spent nuclear fuel for disposal. This may be necessary because foreign research reactor spent nuclear fuel may not be accepted in a geologic repository without some form of chemical processing or treatment. The repository acceptance criteria will not be final until a repository has been licensed.

Any new facilities would be designed to meet modern standards. The new design would minimize air and water emissions and the public and worker radiation doses at least as well as existing facilities, so DOE and the Department of State expect these impacts would be somewhat lower than those presented above for the conventional chemical separation technologies.

Some rough quantitative estimates are possible on the number of canisters that would be produced by some of the developmental technologies for disposal. Table 4-56 compares these estimates to the number of canisters that would be generated by chemical separation. The estimates of numbers of canisters that would be generated by the developmental treatment and/or packaging technologies do not depend on which DOE site performs the treatment and/or packaging.

Table 4-56 Comparison of Geologic Disposal Canisters for Various Technologies

Technology	*Approximate Number of Canisters*
Conventional Chemical Separation	
at the Savannah River Site	72
at the Idaho National Engineering Laboratory	90
Developmental Packaging Technologies	
Direct Disposal in Small Packages	140
Can-in-Canister	240
Developmental Treatment Technologies	
Melt and Poison	25
Chop and Poison	25
Melt and Dilute	180
Dissolve and Poison	950
Chop and Dilute	4,900
Dissolve and Dilute	11,800

The can-in-canister concept was recently introduced (Leventhal and Lyman, 1995), but it could be possible to implement it quickly at the Savannah River Site. Most of the foreign research reactor spent nuclear fuel elements would fit in cans of approximately 10 cm diameter and 85 cm length. If all of the approximately 22,700 elements were placed in these cans, the total canned volume would be about 150 m^3. Using the can-in-canister technology, this volume of glass would be displaced from high-level waste canisters to be produced in the Defense Waste Processing Facility. Since each canister has an internal volume of 0.625 m^3, displacing 150 m^3 of glass would require the production of approximately 240 additional high-level waste glass canisters at the Defense Waste Processing Facility.

The rest of the estimates of numbers of canisters that would be generated by the developmental technologies are scaled from a study (WSRC, 1994a) of the disposition of 7.3 MTHM of aluminum-based spent nuclear fuel, up to the 18.2 MTHM of aluminum-based foreign research reactor spent nuclear fuel. The melt and poison or chop and poison technologies could produce the fewest canisters, as low as 25 canisters. The consolidate and poison technology could produce the next lowest number of canisters (about 140) among the developmental technologies analyzed. The can-in-canister, melt and dilute, dissolve and poison, chop and dilute, and dissolve and dilute technologies would produce increasing numbers of canisters, in that order. The most canisters would be produced by the dissolve and dilute technology: over 11,000 canisters. This uncertainty in the number of canisters translates into a large uncertainty in the cost of disposal. Furthermore, it is not clear which, if any, of these waste forms would be acceptable in a geologic repository.

4.4 Management Alternative 2: Facilitate the Management of Foreign Research Reactor Spent Nuclear Fuel Overseas

The basic implementation of Management Alternative 1 of the proposed action and the seven implementation alternatives to the basic implementation of Management Alternative 1 are all based on acceptance of foreign research reactor spent nuclear fuel into the United States. As discussed in Chapter 2, the two subalternatives under Management Alternative 2 facilitate overseas management of foreign research reactor spent nuclear fuel. This section discusses their policy considerations and environmental impacts. For convenience, the two subalternatives under Management Alternative 2 are defined briefly below:

1. Subalternative 1a - Overseas storage of the foreign research reactor spent nuclear fuel with U.S. technical and/or financial assistance, and

2. Subalternative 1b - Overseas reprocessing of the foreign research reactor spent nuclear fuel with U.S. nontechnical assistance.

Under these subalternatives, no foreign research reactor spent nuclear fuel would be accepted into the United States. The United States would negotiate some form of technical assistance and/or financial incentives in return for maintaining some measure of control over the spent nuclear fuel containing uranium enriched in the United States.

4.4.1 Subalternative 1a: Overseas Storage with U.S. Assistance

Policy Considerations

The foreign research reactor spent nuclear fuel could remain in interim storage overseas. The number of foreign research reactor spent nuclear fuel management sites involved would be greater and the quality of storage technology in some countries might be lower than if the basic implementation of Management Alternative 1, or one of its seven implementation alternatives, was adopted.

The cost of this subalternative might be greater than the cost of the basic implementation of Management Alternative 1 because it might not take advantage of economies of scale. To set up a secure area and a nuclear material handling infrastructure, purchase a storage cask, transfer the spent nuclear fuel to the cask, and maintain the secure area and nuclear infrastructure for 40 years would cost tens of millions of dollars. To repeat this in several dozen countries could potentially push the total cost up into the range of hundreds of millions of dollars. Furthermore, after incurring this expense, all of the U.S. origin HEU would still be located in foreign countries where a change in government could reverse any commitment to withhold the material from production of nuclear weapons.

This subalternative would be economically attractive only in countries that already have nuclear infrastructures. In these cases, the addition of the spent nuclear fuel from research reactors to existing spent nuclear fuel inventories in storage would involve only incremental costs without all the startup costs.

If the United States does not accept any near term foreign research reactor spent nuclear fuel shipments, provision of U.S. technical and/or financial assistance for the development of safe and secure storage capabilities would help to alleviate some of the problems posed by a lack of sufficient storage capacity. However, this subalternative presents several drawbacks from a nuclear weapons nonproliferation policy standpoint. The accumulation overseas of ever larger amounts of spent nuclear fuel containing HEU poses a risk that such weapons-usable material might be illicitly diverted to a weapons program. Although U.S. assistance in maintaining adequate physical security for foreign research reactor spent nuclear fuel repositories may lessen the potential for diversion, the proliferation risks would still be greater than under the basic implementation of Management Alternative 1. As the foreign research reactor spent nuclear fuel ages, it would become less radioactive and thus a more attractive target for illicit diversion.

For countries that will not allow the indefinite storage in their territories of increasing quantities of spent nuclear fuel, this subalternative is not a viable option. Under this scenario, reactor operators in these countries, in order to avoid shutting down, might be forced to consider storing their spent nuclear fuel in other countries, where safe and secure management and material accountancy problems could exist and the risk of illicit diversion could be a concern. For example, Austria was reportedly approached by commercial interests from Belarus with an offer to store spent nuclear fuel from the ASTRA reactor for hard currency. (Since the "Offsite Fuels Policy" for HEU spent nuclear fuel expired in 1988, the Austrian government has required that for fresh fuel to enter the country, an equivalent quantity of spent nuclear fuel must be shipped out of the country.) The offer, which was rejected in support of nuclear weapons nonproliferation policies, is indicative of the scenarios that may develop as pressure builds on reactor operators to close the back end of their nuclear fuel cycle.

Impacts

There would be no environmental impacts on U.S. territory for the duration of the interim period.

4.4.2 Subalternative 1b: Overseas Reprocessing with United States Non-Technical Assistance

4.4.2.1 Overview and Policy Considerations

Foreign research reactor spent nuclear fuel could be reprocessed in foreign facilities and the resulting high-level waste vitrified or cemented. No U.S. reprocessing technology would be used in this subalternative. The inventory and conditions for management of foreign research reactor spent nuclear fuel under Subalternative 1b are the same as those under basic implementation of Management Alternative 1. The amount of HEU that would be removed from international commerce is the same as under basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons)]. To be consistent with U.S. nuclear weapons nonproliferation policy, however, bilateral agreements would have to be established with one or more foreign governments before DOE and the Department of State could consider implementation of such a subalternative.

The advantages and disadvantages of the technology used for reprocessing overseas would be essentially the same as those described for chemical separation in the United States as discussed in Section 2.2.2.6.

There are four sites in Europe at which reprocessing is conducted for commercial customers: the Marcoule and La Hague sites in France, and the Dounreay and Sellafield sites in the United Kingdom. The companies that operate these sites are strictly regulated by their government agencies. The facilities at La Hague and Sellafield are dedicated to oxide spent nuclear fuel from commercial reactors and are not likely candidates for reprocessing the metallic foreign research reactor spent nuclear fuel. All four of these sites routinely release small quantities of radionuclides into the environment and produce radioactive wastes. For example, in 1993 the releases from the Dounreay facility to the North Sea included 2.7 Ci of total alpha activity, 220 Ci of beta activity excluding tritium, and 27 Ci of beta activity from tritium. These releases represented 13 percent, 7.2 percent, and 0.8 percent of the applicable regulatory limits (Jones et al., 1994). The radionuclides released into the atmosphere and into a river or sea would flow across international boundaries. These releases would cause a small, unmeasurable increase in world-wide natural background radiation levels. The transport of vitrified high-level waste away from the reprocessing facility would also produce environmental impacts on foreign territory and possibly in international waters.

Since the United States does not encourage the development of reprocessing capabilities overseas, DOE and the Department of State would only consider this subalternative in France or the United Kingdom where the capability already exists. Reprocessing would most likely take place (as it already has in several instances) at the Dounreay facility—the sole facility currently willing and able to reprocess foreign research reactor spent nuclear fuel. France's facility in Marcoule does reprocess spent nuclear fuel from French research reactors, but does not currently accept such spent nuclear fuel from other nations for reprocessing.

The British and French regulatory agencies require the customer to accept the wastes as a condition of reprocessing spent nuclear fuel, so this option would be unavailable to those countries lacking the technical or legal capability to store or dispose of high-level waste. Alternatively, the United States might consider accepting the wastes from reprocessing.

4.4.2.2 Waste Generation at the Foreign Reprocessing Site

Reprocessing the foreign research reactor spent nuclear fuel would produce two distinct streams: the uranium and the waste products.

For spent nuclear fuel containing HEU, the HEU would be blended down to LEU at the reprocessing facility. If the LEU were then shipped to the United States, the resulting environmental impacts would be no greater than for ordinary nonhazardous cargo because LEU produces such a small radiation dose rate.

The British and French have decades of experience in conditioning nuclear waste at their four reprocessing facilities. In recent years, they have greatly reduced the volumes of wastes that require disposal. Both nations use the same technology for vitrifying their high-level waste, and both nations produce the same size high-level waste glass canister: 0.15 m^3 (5.3 ft^3). These canisters of high-level waste glass are expected to be suitable for disposal in geologic repositories. As of September 1993, France and the United Kingdom had filled more than 2,100 and 350 canisters with high-level waste glass, respectively (Masson, et al., 1994).

As a general rule, European reprocessing and vitrification of about 8 to 10 MTHM of spent nuclear fuel would generate about 1 m^3 (35.3 ft^3) of high-level waste in glass form (UKAEA, 1994; Masson, et al., 1994). Thus, if all 19.2 MTHM of the foreign research reactor spent nuclear fuel were reprocessed and vitrified overseas, DOE and the Department of State estimate that the total volume of vitrified high-level waste would be only about 2.4 m^3 (85 ft^3). DOE and the Department of State estimate that the high-level waste from reprocessing all the foreign research reactor spent nuclear fuel would fill about 16 European-sized canisters. For reference, this volume of glass waste would fill four American-sized canisters.

4.4.2.3 Removal of Waste from the Reprocessing Site(s)

The British and French governments do not accept responsibility for ultimate disposal of the high-level waste glass canisters for foreign customers. Both nations require that disposal of the high-level waste glass canisters and any other wastes generated during reprocessing of their spent nuclear fuel, including low-level waste, be the responsibility of the nation(s) hosting the reactors. At the Dounreay Site, however, only small amounts of low-level waste have been generated during reprocessing of spent nuclear fuel from research reactors. Many nations with foreign research reactors, however, do not have any capabilities to accept the high-level waste glass canisters. The United States may accept the intact foreign research reactor spent nuclear fuel from these nations while simultaneously encouraging the nations which can accept the canisters to reprocess their foreign research reactor spent nuclear fuel under the conditions noted in Section 4.4.2.1. This would be a combination of the basic implementation of Management Alternative 1 and Subalternative 1b (overseas reprocessing) of Management Alternative 2.

As another option under this subalternative, if the host nations cannot accept this responsibility, the United States would commit to accept the high-level waste glass canisters. This could provide the incentive necessary to convince reactor operators to cooperate with the RERTR program and to use LEU in their reactors. Some nations may refuse to reprocess or require the United States to take title to the foreign research reactor spent nuclear fuel prior to reprocessing.

DOE and the Department of State could begin accepting canisters into the United States within the first 10 years, or DOE and the Department of State could specify that they be stored at the reprocessing facility for decades. If the canisters were accepted in the near term, they would most likely be stored at the Savannah River Site because this site has already built a new storage facility with a capacity of 2,286 canisters. If the canisters were stored overseas for decades, then they would be transported directly to the geologic repository.

Marcoule produces vitrified waste, similar to U.S. vitrified waste. In the United Kingdom on the other hand, as a result of a different regulatory structure, the wastes from reprocessing of research reactor spent nuclear fuel are classified as intermediate-level radioactive wastes. (In the United States, these same materials would be classified as high-level radioactive wastes.) In the United Kingdom, the intermediate-level wastes are mixed with a special cement and poured into steel drums, which can then be buried. This waste form is dissimilar to the vitrified borosilicate glass high-level waste form that is expected to be produced in the United States, and is incompatible with United States radioactive waste disposal standards. The government of the United Kingdom might allow an exchange of vitrified commercial waste from Sellafield for cemented waste from Dounreay, which might allow the United States to accept vitrified high-level waste from the United Kingdom.

Transportation of vitrified high-level waste must conform to U.S. Department of Transportation (49 CFR Part 173) and NRC (10 CFR 71) regulations. Under this option, the European-sized glass canisters would be transported in "Type B" casks, which provide a high degree of assurance that cask integrity will be maintained with essentially no loss of radioactive contents or serious impairment of the shielding capability provided by the cask, even in severe accidents. DOE has prepared initial designs for a defense high-level waste cask for truck transportation of the Savannah River Site high-level waste. As initially designed, the defense high-level waste cask uses a solid body concept to absorb energy during an accident and normal transportation conditions. To minimize the exposure to gamma radiation, shielding would be provided by a depleted uranium liner inside the cask body. (Gamma radiation is high-energy, short wavelength electromagnetic radiation with properties similar to x-rays.) The regulatory limit for radiation dose rate outside the cask is 10 mrem per hour at 2 m (6.6 ft) from the edge of the vehicle. Casks transported under this option are assumed to emit this level of radiation. Currently, however, no casks for shipping high-level waste canisters by truck or rail have been certified by the NRC.

Each of these "Type B" casks would be large enough to hold two European-sized glass canisters. Thus, the option of overseas reprocessing with acceptance of approximately 16 high-level waste glass canisters would require about 8 cask shipments into the United States (versus 721 cask shipments by sea and 116 by land under the basic implementation of Management Alternative 1). Vitrified high-level waste shipments would use the same East Coast port(s) identified in Chapter 2 for foreign research reactor spent nuclear fuel. The same procedures and representative overland routes analyzed for foreign research reactor spent nuclear fuel would apply to these shipments of vitrified high-level waste. The management site for these canisters would be the Savannah River Site. Alternatively, they might be transported directly to the candidate geologic repository at Yucca Mountain, NV.

Each of the eight casks is assumed to contain the waste products associated with one-eighth of the foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1.

Marine Transport Impacts

Risks under Subalternative 1b were assessed using the same methodology used to evaluate risks associated with the transport of the foreign research reactor spent nuclear fuel. The major differences in the analysis are the number of cask shipments and the isotopic content within each transportation cask.

Impacts of Incident-Free Marine Transport

As with the shipment of foreign research reactor spent nuclear fuel, the primary impact of incident-free marine shipping of vitrified waste would be upon the crews of the ships. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste. The primary contribution to the crew dose would come from the daily cargo inspection activities. Three crew members have been modeled as performing the inspections and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis it has been assumed that the vitrified waste would be transported on a chartered vessel, there would be no intermediate port calls, and the shipment would originate in Europe (either the United Kingdom or France.)

As in the spent nuclear fuel analysis, either two or eight casks are assumed to be on each single voyage. This assumption results in exposure to two radiation fields during all activities that bring crew members into the vicinity of the transportation casks. Should all the casks be shipped at once, this assumption is equivalent to assuming that this single voyage is made with two casks per hold in one vessel. The crew risk would be the same for this single voyage as for four voyages with two casks per vessel.

Results of the marine incident-free risk analysis are presented in Table 4-57. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the crew would be approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, would be well below the DOE and NRC limits for public exposure of 100 mrem per year. If, however, all the casks were shipped in 1 year (perhaps all on one ship), then the maximally exposed worker dose would exceed the limit of 100 mrem per year. In this case, new inspectors would be used to keep each individual's dose below the limit.

Table 4-57 Incident-Free Marine Transport Impacts (Subalternative 1b)

	Maximally Exposed Worker Dose (mrem)	*Maximally Exposed Worker Risk (LCF)*	*Population Dose to Crew (person-rem)*	*Population Crew Risk (LCF)*
Per voyage (2 casks)	53	0.000021	0.19	0.00007
Entire program	210	0.000084	0.74	0.00030

Impacts of Accidents During Marine Transport

If the ship carrying a cask of vitrified waste were to catch fire at sea and the cask was sufficiently damaged by fire to release its contents, members of the ship's crew near the fire would be exposed to the released radioactive material. Any resulting plume carrying radioactive particles would disperse over the ocean, where there is no human population. Therefore, the ship's crew would be the only people exposed to the released radioactive material. The number of ship's crew members is considerably smaller than the population modeled within a short distance of an accident that occurs in the port. Therefore, consequences of a shipboard accident resulting in the release of radioactive material in a plume would be covered by the consequences of the accidents considered in the port analysis. As discussed below, because the oceans are a very dilute system, effects on marine biota would not be discernible.

If a collision or other accident (e.g., loss of a cask over the side in a storm) occurred in which an intact cask fell overboard, the fact that the cask would be immersed would not necessarily result in a release of its contents. Spent nuclear fuel casks are designed to withstand at least a 15-m (50-ft) immersion, and it has been demonstrated that the cask seals will remain intact at much greater depths (IAEA, 1990). Spent nuclear fuel casks, damaged or undamaged, can be recovered from water up to 200 m (660 ft) deep: well beyond the range typical of coastal and port depths. (Recovery at great depths, e.g., more than 2,000 m or 6,600 ft, is possible, but would be costly). It is reasonable to believe that a cask would be recovered in any incident involving the immersion of a cask in waters up to 200 m (660 ft) in depth.

The Nuclear Energy Agency of the Organization for Economic Cooperation and Development, Paris, France, estimated the impacts of various accident scenarios involving shipment of reprocessed commercial spent nuclear fuel. The Nuclear Energy Agency estimated that a damaged and unrecovered cask of high-level waste in coastal waters would result in a peak individual human dose of 6.5 mrem per year per MTHM (NEA, 1988). Dose and exposure estimates that follow are based on the estimates generated in the Nuclear Energy Agency study and are modified to take into account the content of the casks based on the shipment of all material from the foreign research reactor spent nuclear fuel program in eight cask shipments.

In the most extreme situation, where the accident occurs in coastal waters, the spent nuclear fuel is not recovered, and the cask is damaged, the peak dose to an individual human is estimated to be 19 mrem per year. The individual is assumed to reside near the shore and ingest seafood (fish, mollusk, and seaweed) harvested from the area in the immediate vicinity of the vitrified waste transportation cask. (For an initially intact cask, the dose would be expected to be considerably lower, approximately 0.3 mrem per year.) Peak biota doses are estimated at 0.8 mrad per year for fish, 0.9 mrad per year for crustaceans, and 19 mrad per year for mollusks, if the cask were damaged and not retrieved from coastal waters. With cask retrieval, both the peak dose to an individual and the biotic impacts would be considerably smaller. The results for the loss and failure of a single cask are lower than the peak impacts for the loss and failure of a single spent nuclear fuel cask (see Section 4.2.1.3), principally due to the lower leach rate for vitrified waste (see Appendix C).

In deep waters, the radioactive constituents of the vitrified waste would be released slowly over time into the surrounding waters if the cask were not recovered. Some of the radioactive material would be removed from the water by adhesion to suspended sediments. Assuming a damaged cask of vitrified high-level waste were submerged on the deep ocean bottom, the peak human individual dose to an individual residing along the coast and ingesting seafood harvested from the general area in which the breached submerged cask is located is estimated to be 0.000015 mrem per year.

Humans would not be the principally exposed species in a deep ocean accident involving vitrified waste casks. Using the Nuclear Energy Agency estimates and assuming that the damaged waste cask lay on the ocean floor where it slowly released its radioactive inventory, the peak doses to biota residing on the ocean floor in or near the uppermost sediment layer would be 0.9 rad per year for fish, 1.2 rad per year for crustaceans, and 41 rad per year for mollusks (NEA, 1988).

Harmful effects of chronic irradiation have not been observed in natural aquatic populations at dose rates less than 365 rad per year (NCRP, 1991). At doses an order of magnitude below this, as would be the case in an accident involving the vitrified waste from the foreign research reactor spent nuclear fuel, it is unlikely that either a population of marine biota or individual members of that population would be harmed by the radiation resulting from a spent nuclear fuel accident. Furthermore, no chemical hazard would be expected from the release of the contents of the vitrified waste canisters into the open ocean.

Using the same accident probabilities used in the marine transport analysis of the basic implementation of Management Alternative 1, risk estimates were developed for this subalternative. The MEI risk due to the loss of a vitrified high-level waste cask in the ocean is very low for the shipment of up to eight casks. The highest estimated risk to a human would occur in the accident scenario in which a cask is sunk and not recovered from coastal waters. This scenario would result in an MEI risk on the order of 1 x 10^{-10} mrem per year, which corresponds to about 2.7 x 10^{-15} LCF. This means that the chance of the MEI incurring one LCF due to this subalternative would be about one in one quadrillion.

Port Activity Impacts

Impacts of Incident-Free Port Activities

As with the shipment of the foreign research reactor spent nuclear fuel, the primary impact of incident-free port activities required to unload the vitrified waste casks is upon the workers: port handlers, inspectors, and port staging personnel. Most of the assumptions used in the analysis of the port worker exposure to the foreign research reactor spent nuclear fuel (see Section 4.2.2.2) have been used to analyze the impact of the shipment of vitrified waste.

Results of the port activities' incident-free risk analysis are presented in Table 4-58. Due to the reduced number of cask shipments, compared to the approximately 721 marine cask shipments of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1, the risks to the port workers are approximately 2 orders of magnitude lower than those calculated in Section 4.2.2.2 for the basic implementation of Management Alternative 1. The doses to the crew, including the maximally exposed worker, are well below the DOE and NRC limits for public exposure of 100 mrem per year.

Impacts of Accidents During Port Activities

The methodology used to evaluate the accident consequences and risks associated with port accidents is identical to that used to assess these items for the shipment of foreign research reactor spent nuclear fuel under the basic implementation of Management Alternative 1 (Section 4.2.2.3). The MACCS code was used with site-specific population and meteorology data to determine the consequences of an accident. The inventory (radionuclide content) of the transportation casks was determined by combining the radionuclide content of all of the vitrified waste to be returned to the United States under this subalternative and equally dividing it among the eight casks. In this analysis it was assumed that the Canadian spent nuclear fuel, which was assumed to be sent to the United States via truck in the analysis documented in Section 4.2.2.3, would be sent to Europe and reprocessed. The vitrified waste from this spent nuclear fuel is included in this analysis.

Table 4-58 Incident-Free Port Activity Impacts (Subalternative 1b)

Impacts Per Shipment				
	Maximally Exposed Worker Dose (mrem)	***Maximally Exposed WorkerRisk (LCF)***	***Population Dose (person-rem)***	***Population Risk (LCF)***
Inspectors	1.3	5.2×10^{-7}	0.0053	0.0000021
Port Handlers	0.46	1.8×10^{-7}	0.0015	0.00000061
Port Staging Personnel	0.38	1.5×10^{-7}	0.0046	0.0000018
Maximum	**1.3**	$\mathbf{5.2 \times 10^{-7}}$	----	----
Total	----	----	**0.011**	**0.0000042**
Impacts for the Entire Subalternative 1b				
	Maximally Exposed WorkerDose (mrem)	***Maximally Exposed Worker Risk (LCF)***	***Population Dose to Workers (person-rem)***	***Population Risk to Workers (LCF)***
Inspectors	10	4.0×10^{-6}	0.04	0.000017
Port Handlers	4	1.6×10^{-6}	0.01	0.0000048
Port Staging Personnel	3	1.2×10^{-6}	0.04	0.000015
Maximum	**10**	$\mathbf{4.0 \times 10^{-6}}$	----	----
Total	----	----	**0.09**	**0.000036**

The amounts of material released from the glass in various accident scenarios, called release fractions, are based on information developed for accident analysis at the Savannah River Site (DOE, 1994k). These release fractions are the same for the three accident categories analyzed for the spent nuclear fuel port accidents (the accident categories included collisions and collisions followed by fires). Therefore, these three accident categories were combined to form a single category for this analysis. Accident probabilities were developed for this single accident category at both the dock and in the approach to the dock.

Since all of the vitrified waste would be transported to the United States from Europe, only East Coast ports were selected for port-specific analysis of the accident consequences. The port accident analysis was performed for three East Coast ports: Philadelphia, PA; Charleston, SC; and MOTSU in North Carolina. These three ports represent a wide range of port city populations. As in the port accident analysis discussed in Section 4.2.2.3, these ports are not necessarily the selected ports of entry for the vitrified waste. They are intended to be representative of the range of populations, and therefore consequences, associated with all of the potential ports of entry.

Results of this analysis are presented in Table 4-59. The consequences of an accident in port involving a cask of vitrified waste would be lower than for the category 5 and 6 accidents involving the foreign research reactor spent nuclear fuel casks. This is a result of the much lower release fractions associated with the vitrified waste compared to the release fractions for the metallic spent nuclear fuel. However, for the vitrified waste, category 4 accidents result in the release of the same amount of material from the vitrified waste as the category 5 and 6 accidents. (For foreign research reactor spent nuclear fuel, category 4 accidents result in much smaller consequences.) Because the frequency of this category of accidents is two orders of magnitude higher than that for category 5 and 6 accidents, the port accident risks per single-cask shipment are higher for vitrified waste than for foreign research reactor spent nuclear fuel. The port accident population risks would be about the same order of magnitude as those under the basic implementation of Management Alternative 1 because of these category 4 accidents.

Table 4-59 Port Accident Risks (Subalternative 1b)

Port	Risk per Single-Cask Shipment of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

The MEI doses calculated for these accidents have a rather small variance. The largest estimated MEI dose is 740 mrem. The largest probability of one LCF (given that the accident has occurred) was 0.00035. Combining these estimates with the probability of a severity category 4 accident per shipment and the number of shipments results in an MEI risk of 1.8×10^{-8} LCF.

Ground Transport Impacts

Under Subalternative 1b, DOE and the Department of State would transport eight casks of vitrified high-level waste overland from an East Coast port(s) to a candidate geologic repository (in Nevada for example). The shipments may go directly from the port(s) to the candidate geologic repository or they might go from the ports to the Savannah River Site for storage, then from the Savannah River Site to the candidate geologic repository. Results are displayed in Figures 4-18 and 4-19.

Impacts of Incident-Free Ground Transport (Ports to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate near vehicles carrying vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. Incident-free transportation of vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00023 to 0.0032 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.0001 to 0.0008. The estimated number of radiation-related LCF for the general population ranged from 0.00009 to 0.0024, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005. The impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

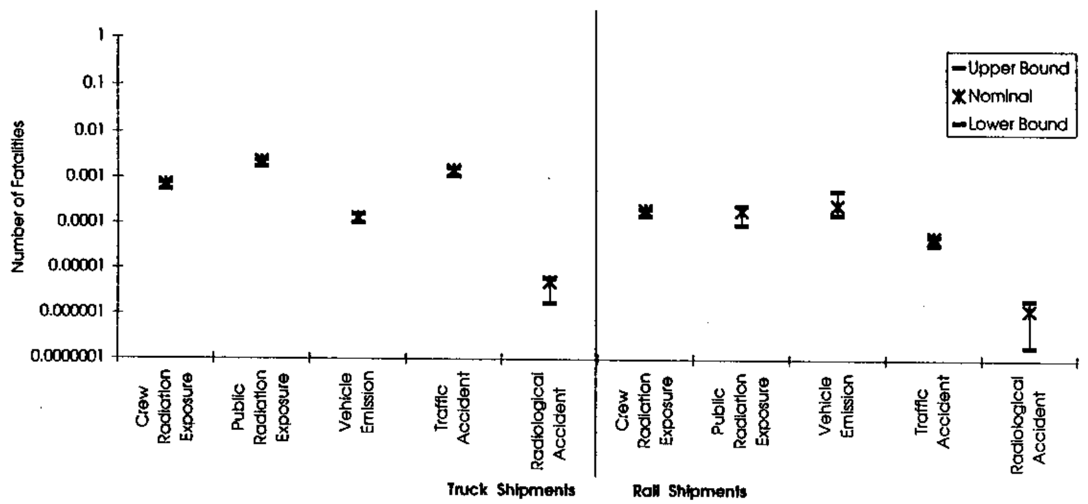
To estimate the maximally exposed ground transport worker risk, DOE and the Department of State assumed all the vitrified waste was transported during a 1-year period and one truck driver received his annual limit of 100 mrem during that year. This dose translates into a risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.0000002 to 0.0000059 LCF from radiation and from 0.00003 to 0.0016 for traffic fatality, depending on the transportation mode and the port(s) selected.

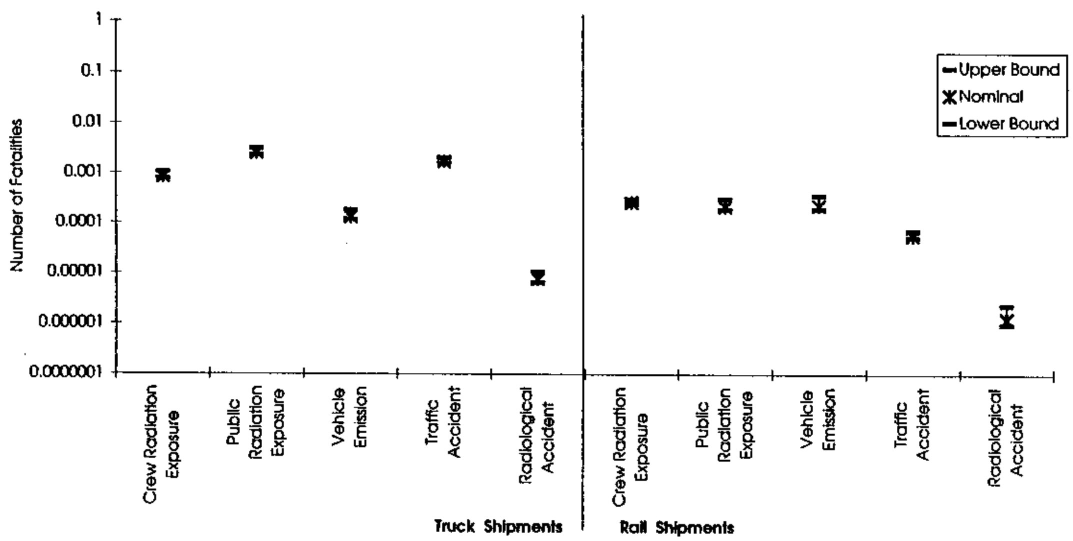
Impacts of Incident-Free Ground Transport (Ports to the Savannah River Site to Repository)

Impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The dose rate from casks containing vitrified waste was assumed to equal the regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the vehicle. The



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-18 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Repository)



- *Traffic Accidents are expressed in immediate fatalities, all other values are latent cancer fatalities*

Figure 4-19 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 2, Subalternative 1b (Ports to Savannah River Site to Repository)

incident-free transportation of the vitrified high-level waste was estimated to result in total latent fatalities that ranged from 0.00041 to 0.004 over the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The estimated number of radiation-related LCF for transportation workers ranged from 0.00023 to 0.001. The estimated number of radiation-related LCF for the general population ranged from 0.00018 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.00011 to 0.00035. Impacts of transportation of vitrified waste canisters are described in more detail in Appendix E.

To estimate the maximally exposed worker risk, it was assumed that the two legs of ground transport would be separated by a long storage period. That is, the second leg (transport from the Savannah River Site to the repository) would occur at least 20 years after the first leg (transport from the ports to the Savannah River Site). Thus, one individual truck driver would probably not be involved in both legs. DOE and the Department of State further assumed that each leg would last no more than 1 year, so no individual truck driver could receive more than the annual regulatory limit of 100 mrem. This translates into a maximally exposed worker risk of 0.00005 LCF.

Impacts of Accidents During Ground Transport (Ports to the Savannah River Site to Repository)

Cumulative transportation accident risks over the vitrified waste shipment program are estimated to range from 0.000001 to 0.00001 LCF from radiation and from 0.00005 to 0.002 for traffic fatality, depending on the transportation mode and the port(s) selected.

The consequences of the maximum foreseeable offsite transportation accident are greater than those of the basic implementation of Management Alternative 1. The frequency, however, is lower due to the reduced amount of ground transport. Maximum estimated MEI risk is reduced to 7×10^{-12} LCF.

Management Site Impacts

Impacts of Incident-Free Management Site Activities

Environmental impacts associated with the receipt and storage of the vitrified high-level waste canisters under Subalternative 1b are limited to the exposure of the working crew that would handle the incoming canisters at the site. The 16 canisters of vitrified waste (approximately 0.15 m^3 or 5.3 ft^3 each) would be received in 8 shipping casks and stored at the Glass Waste Storage Building at the Savannah River Site. The facility, described in Appendix F, has been designed for vitrified waste and has space for 2,286 canisters. Vitrification of all existing liquid high-level waste at the Savannah River Site is expected to produce a total of approximately 5,717 canisters. The impact of this additional amount of glass waste on the operational characteristics of the facility would be very low.

Vitrified waste would not contain any gaseous fission products, so there is no mechanism for incident-free emissions of radioactive material. Thus, impacts to the public near the Savannah River Site under this subalternative would be equal to zero.

To estimate the maximally exposed worker dose, DOE and the Department of State assumed that all the canisters would be received during one year. This is reasonable because of the small number of cask shipments. Then DOE and the Department of State conservatively assumed that one of the workers involved in handling these shipments would receive the maximum annual dose of 5,000 mrem allowed by regulation. This dose translates into an increased risk of 0.002 LCF.

The population dose to workers handling the eight casks would be 2.6 person-rem, based on the methodology presented in Appendix F, Section F.5 for unloading and storing in a vault-type dry storage structure. This translates into a worker population risk of 0.001 LCF.

Impacts of Accidents Onsite

The addition of 16 European-sized canisters to the thousands of larger American-sized canisters is expected to increase the accident risk by a very small increment, so this increase in the risk was not specifically analyzed in this EIS. The accident analysis for the Defense Waste Processing Facility has been reported in its Final EIS (DOE, 1994e).

Since vitrified waste contains no gaseous fission products, however, it is clear that the spent nuclear fuel element breach accident scenarios are not applicable to this subalternative. Thus, the aircraft-crash-with-fire scenario would present the highest risks. The highest annual estimates of MEI/NPAI and population risks under the basic implementation of Management Alternative 1 for this accident scenario are 1.2×10^{-9} LCF and 0.0000015 LCF, respectively (see Section 4.2.4.1). DOE and the Department of State consider these estimates to cover the risks for vitrified waste because the vitrified waste is designed to be much more stable than spent nuclear fuel in all accidents. Multiplying these annual estimates by the number of years the accident might occur (30 years) yields the risks for this alternative: 3.6×10^{-8} LCF for the MEI/NPAI risk and 0.000045 LCF for the population risk.

4.4.2.4 Disposal Site Impacts

Whether the vitrified high-level waste canisters were managed at the Savannah River Site or in Europe, eventually they would be transported to a geologic repository for disposal under this subalternative. Current planning for the U.S. candidate geologic repository at Yucca Mountain in Nevada indicates that acceptance of high-level waste canisters would begin early enough that the high-level waste from foreign research reactor spent nuclear fuel could be shipped to and emplaced in the repository before the end of the interim period.

Impacts due to handling European-sized canisters at the repository would be similar to the impacts due to handling American-sized canisters. After emplacement in the disposal site, no more impacts are expected to workers, the public, or the environment for at least 10,000 years because the radioactive material would be extremely unlikely to escape from the repository.

4.4.2.5 Summary of the Impacts of Subalternative 1b

The principal impacts under Subalternative 1b would be occupational and public health and safety impacts. These impacts would be due to the acceptance of vitrified high-level waste into the United States from Europe. (If no high-level waste were accepted, then there would be no impacts on U.S. territory.) These impacts are presented in Table 4-60 in terms of the risk of death due to cancer during each of the four segments of the affected environment. It also shows, in the bottom rows, the highest of the individual risks and the total population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population.

Table 4-60 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of high-level waste producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing

Table 4-60 Maximum Estimated Radiological Health Impacts of Subalternative 1b

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.000084	0	0.0003
Accidents	2.7×10^{-15}	much less than 0.00002	---
Port Activities			
Incident-Free	0.000004	0	0.000036
Accidents	1.8×10^{-8}	0.00002	---
Ground Transport			
Incident-Free	0.00005	0.003	0.001
Accidents	7×10^{-12}	0.00001	---
Site Activities			
Incident-Free	0.002	0	0.001
Accidents	3.6×10^{-8}	0.000045	---
Highest Individual Risk			
Incident-Free	0.002	----	----
Accidents	3.6×10^{-8}	----	----
Total Population Risk			
Incident-Free	----	0.003	0.0027
Accidents	----	0.000075	----

people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE site receiving the maximum dose allowed by DOE regulation (5,000 mrem) during the 1 year of high-level waste acceptance.

The highest estimated incident-free individual risk is 0.002 LCF, which would apply to an onsite radiation worker. This individual would have a one in five hundred chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally worker risk. DOE estimates this risk to be very nearly zero LCF.

The maximum estimated accident MEI risk is 3.6×10^{-8} LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten million. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

The total incident-free population risk for both the general public and workers would be much less than one LCF.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-60. There is about a 0.2 percent chance that a truck driver or member of the public could die in a traffic accident associated with this subalternative. This death would be unrelated to the radioactive nature of the cargo.

4.5 Management Alternative 3 - Combination of Elements from Management Alternatives 1 and 2 (Hybrid Alternative)

As discussed in Section 2.4, DOE and the Department of State could combine implementation elements from Management Alternatives 1 and 2. Analysis of this example Hybrid Alternative does not signify its preference over other possible Hybrid Alternatives.

Under this Hybrid Alternative, DOE and the Department of State would facilitate reprocessing of the foreign research reactor spent nuclear fuel at western European reprocessing facilities (i.e., Dounreay or Marcoule), as in Management Alternative 2. It is assumed that the foreign research reactor operators in countries that can accept the reprocessing waste would agree to this arrangement. DOE would accept and manage the remaining foreign research reactor spent nuclear fuel in the United States as in Management Alternative 1. (Refer to Section 2.4 for a more detailed description of this Hybrid Alternative).

Based on the current capabilities of overseas reprocessors, and for purposes of this analysis, only aluminum-based foreign research reactor spent nuclear fuel is assumed to be considered for reprocessing; all TRIGA spent nuclear fuel is assumed to be stored in the United States.

Under the Hybrid Alternative, the aluminum-based foreign research reactor spent nuclear fuel to be managed in the United States would be chemically separated at the Savannah River Site as in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States), discussed in Sections 2.2.2.6 and 4.3.6. The uranium and waste products from this chemical separation would be managed as described in Sections 2.2.2.6 and 4.3.6, and the impacts of these activities would be covered by the impacts presented in those sections. The TRIGA spent nuclear fuel would be transported to the Idaho National Engineering Laboratory where it would be stored at existing storage facilities until ultimate disposition. This distribution of the spent nuclear fuel is consistent with the Programmatic SNF&INEL Final EIS (DOE, 1995c) Regionalization by Fuel Type alternative.

The environmental impacts associated with the foreign research reactor spent nuclear fuel that would be accepted into the United States, and the policy considerations of the Hybrid Alternative, are discussed below.

Policy Considerations

Under the Hybrid Alternative, up to 5.3 MTHM and about 5,600 elements of foreign research reactor spent nuclear fuel would be reprocessed overseas. The rest of the foreign research reactor spent nuclear fuel included in the basic implementation of Management Alternative 1, up to 13.9 MTHM and about 17,100 elements, would be accepted into the United States. Overall, the same amount of HEU as in the basic implementation of Management Alternative 1 would be removed from international commerce, up to about 4.6 metric tons (5.1 tons) of HEU.

4.5.1 Marine Transport Impacts

Impacts of Incident-Free Marine Transport

Impacts of incident-free marine transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. Incident-free transportation of spent nuclear fuel was estimated to result in total LCF that ranged from 0.021 to 0.024 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the ships' crews.

The range of impacts results from the analysis of shipment of the spent nuclear fuel on regularly scheduled commercial breakbulk vessels and on chartered container vessels and would be the same as for vessels analyzed under the basic implementation of Management Alternative 1. As in the basic implementation of Management Alternative 1, the difference between the two estimates is a result of the shorter vessel journey time for chartered vessels due to the intermediate port stops associated with the regularly scheduled commercial transport of the spent nuclear fuel.

The highest estimate of the incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF for all the shipments combined).

Impacts of Accidents During Marine Transport

Population risks due to accidents under the Hybrid Alternative would be reduced from those associated with the basic implementation of Management Alternative 1 because of the reduced amount of marine transport. As before, the population risks of accidents at sea are bounded by the risk of accidents in port.

The maximum consequences of the at-sea accidents for the Hybrid Alternative are no different than those of at-sea accidents associated with the basic implementation of Management Alternative 1. For an accident involving the loss of a transportation cask in coastal waters, the maximum exposure to an individual is estimated to be 14,000 mrem per year. DOE and the Department of State would mitigate this impact, however, by recovering the cask. Due to the reduced number of cask shipments compared to the basic implementation of Management Alternative 1, the likelihood of such an accident would also be reduced. The Hybrid Alternative would require approximately 63 percent of the number of shipments required under the basic implementation of Management Alternative 1. The highest estimated risk due to an accident during marine transport would therefore be 0.00012 mrem per year peak dose to a human from the loss of a damaged cask in the deep ocean. This corresponds to an MEI risk of about 3×10^{-10} LCF. This means that this individual would have a chance of less than one in a billion of incurring an LCF due to an accident during marine transport.

4.5.2 Port Activity Impacts

Impacts of Incident-Free Port Activities

In the analysis of the basic implementation of Management Alternative 1, the radiological impact of port activities was estimated on a per-shipment basis. The Hybrid Alternative would require about 63 percent of the number of cask shipments required under the basic implementation of Management Alternative 1. The incident-free impacts of the port activities are proportionally reduced. The estimated number of LCF associated with this alternative range from 0.0021 to 0.0076. As in the marine incident-free analysis, this range of impacts is the result of the analysis of two modes of spent nuclear fuel shipment, regularly scheduled commercial breakbulk vessels and chartered container vessels.

The highest estimate of incident-free maximally exposed worker risk is the same as for the basic implementation of Management Alternative 1 (0.00052 LCF).

Impacts of Accidents During Port Activities

Port accident risks were calculated based on the per-shipment risks determined in the analysis of the basic implementation of Management Alternative 1. The analysis examined the impact of using a wide range of ports based on the population around the port city, from high density population ports such as Elizabeth, NJ, to low-density ports such as the MOTSU terminal in North Carolina. The analysis also considered the impact of chartered shipments (no intermediate port stops before the vessel reaches the spent nuclear fuel port of entry) versus regularly scheduled commercial shipments with up to two intermediate ports of call before the spent nuclear fuel port of entry. Port accident risks associated with the Hybrid Alternative are estimated to range from 2×10^{-7} to 0.00002 LCF from radiation. The range of fatality estimates is due to both the differences in port city populations and the number of intermediate port stops.

Consequences of the maximum foreseeable port accident are identical to those of the basic implementation of Management Alternative 1. The frequency is lower due to the reduced number of cask shipments, so the MEI risk is reduced to about 1 x 10^{-10} LCF.

4.5.3 Ground Transport Impacts

Impacts of Incident-Free Ground Transport

Radiological impacts of incident-free ground transportation were analyzed in the same manner as for the basic implementation of Management Alternative 1. The results are presented in Figure 4-20. The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.011 to 0.15 over the 13-year duration of the acceptance program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crew.

The range of fatality estimates is caused by two factors: the option of using truck or rail to transport spent nuclear fuel and the possibility of using different ports that created varying shipment distances.

The estimated number of radiation-related LCF for transportation workers ranged from 0.008 to 0.037. The estimated number of radiation-related LCF for the general population ranged from 0.010 to 0.11, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0031 to 0.025. Since these risk numbers are much less than one, implementation of the Hybrid Alternative would be unlikely to result in one LCF.

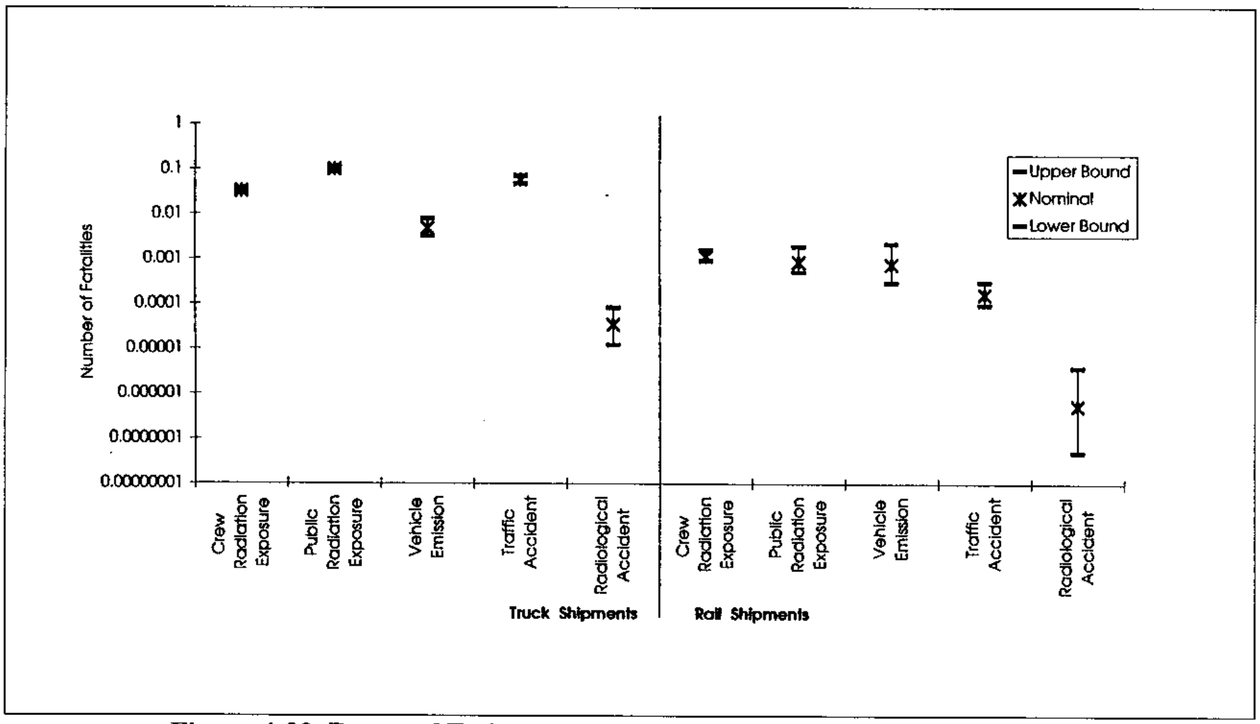


Figure 4-20 Range of Estimated Fatalities (Latent and Immediate) Under Management Alternative 3 (the Hybrid Alternative)

Impacts of Accidents During Ground Transport

Transportation accident population risks over the entire Hybrid Alternative are estimated to range from 0.000005 to 0.000081 LCF from radiation and from 0.002 to 0.069 for traffic fatality, depending on the transportation mode and the ports that might be selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation.

The maximum foreseeable offsite transportation accident is identical to that for the basic implementation of Management Alternative 1. The risk is reduced to 7.1×10^{-12} LCF due to the reduced amount of ground transport.

4.5.4 Management Site Impacts

Under the Hybrid Alternative, the amount of foreign research reactor spent nuclear fuel that would be accepted into the United States is about 17,100 elements and 13.9 MTHM. All the TRIGA spent nuclear fuel, representing approximately 4,900 elements and 1.0 MTHM, would be received and stored in existing facilities at the Idaho National Engineering Laboratory. Aluminum-based spent nuclear fuel, representing approximately 12,200 elements and 12.9 MTHM, would be received and chemically separated at the Savannah River Site as described in Implementation Alternative 6 to Management Alternative 1 (near term chemical separation in the United States). Environmental impacts associated with the receipt and storage of the TRIGA spent nuclear fuel at existing facilities at the Idaho National Engineering Laboratory would be covered by the impacts presented for the basic implementation of Management Alternative 1 without construction of new facilities (Section 4.2). Environmental impacts associated with the receipt and chemical separation of the aluminum-based spent nuclear fuel at the Savannah River Site would be covered by the impacts presented for the near-term chemical separation alternative at the Savannah River Site (Section 4.3.6). The occupational and public health and safety impacts for both sites were estimated by combining the appropriate results from earlier analyses for the Idaho National Engineering Laboratory and the Savannah River Site.

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 elements that would be received and managed at the Idaho National Engineering Laboratory under this alternative represent about 22 percent of the number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the Hybrid Alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Multiplying the results in Table 4-9 by the maximum duration of each activity (13 years for receipt and 40 years for storage) yields the highest estimated risks for this part of the Hybrid Alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

The approximately 12,200 elements that would be received at the Savannah River Site under this alternative represent about 54 percent of the number of elements that would be received and temporarily stored there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions during receipt at the Savannah River Site under the basic implementation of Management Alternative 1 are presented in Table 4-8. The impacts for storage in RBOF are much smaller than those for receipt. Multiplying these results by 54 percent and the maximum duration of 13 years yields the highest estimated risks for this part of the Hybrid Alternative. The highest estimated public MEI risk is 3.9 x 10^{-10} LCF and the corresponding estimated public population risk is 0.000020 LCF.

The approximately 12.9 MTHM that would be chemically separated at the Savannah River Site under this alternative represents about 71 percent of the MTHM that would be chemically separated there under Implementation Alternative 6 dedicated to foreign research reactor spent nuclear fuel. Public impacts due to this implementation alternative were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated impacts to the public near the Savannah River Site due to this part of the Hybrid Alternative. Using this procedure, the highest estimated incident-free public MEI risk at the Savannah River Site is 0.0000031 LCF. The highest estimated incident-free public population risk at the Savannah River Site (including both the air and water exposure pathways) is 0.13 LCF.

The maximum of the three onsite activities' estimated public incident-free MEI risks is equal to 0.0000031 LCF, which would result from chemical separation activities at the Savannah River Site (The three parts are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to the Hybrid Alternative would be less than one in one hundred thousand.

The total of the three onsite activities' estimated public incident-free population risks is 0.13 LCF.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of this Hybrid Alternative is 13 years, the same as that in both the basic implementation of Management Alternative 1 and Implementation Alternative 6. Thus, the estimated maximally exposed worker dose is also the same. The maximally exposed worker risk is estimated to be 0.026 LCF.

Incident-free worker population impacts due to the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4. Using the same evaluation process described in Appendix F, Section F.5, for the 162 casks of TRIGA foreign research reactor spent nuclear fuel that would be received and unloaded under this Hybrid Alternative yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the Hybrid Alternative is 0.021 LCF.

Workers at the Savannah River Site would receive and unload 406 casks of aluminum-based foreign research reactor spent nuclear fuel in an existing wet facility under this alternative, receiving a population dose of 157 person-rem. The associated worker population risk for this part of the Hybrid Alternative is 0.063 LCF.

Incident-free worker population impacts due to Implementation Alternative 6 (chemical separation) were presented earlier in this chapter in Table 4-48. Multiplying these results by 71 percent yields the estimated incident-free impacts to the workers at the Savannah River Site due to the Hybrid Alternative. Using this procedure, the highest estimated incident-free worker population risk due to chemically separating this spent nuclear fuel at the Savannah River Site is 0.078 LCF.

The total of the three onsite activities' estimated incident-free worker population risks is 0.16 LCF.

Impacts of Accidents Onsite

Accident scenarios, frequencies, consequences, and annual risks for the Hybrid Alternative are derived from those for the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory and Implementation Alternative 6 at the Savannah River Site.

Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory were presented earlier in this chapter in Table 4-25. Multiplying these by the duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory under this alternative. The receipt/unloading impacts are reduced by the factor of 22 percent. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the mixture of TRIGA and aluminum-based spent nuclear fuel in the basic implementation of Management Alternative 1. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 1.9×10^{-6} LCF, which is due to an accidental criticality in a wet storage facility. The highest estimated accident population risk for this part of the Hybrid Alternative is 0.0088 LCF, which is due to the same accident scenario.

Annual accident risks for receipt and unloading at the Savannah River Site were presented in Table 4-24. Multiplying these by the duration of the receipt activity (13 years) yields the risk due to receipt and temporary storage accidents under this alternative. The highest estimated accident MEI risk for this part of the Hybrid Alternative is 2.6×10^{-6} LCF, which is due to an accidental criticality in RBOF. The corresponding estimated accident population risk for this part of the Hybrid Alternative is 0.096 LCF.

The accident MEI and population impacts due to chemical separation were presented earlier in this chapter in Table 4-50. Multiplying these results by 71 percent yields the estimated impacts at the Savannah River Site due to accidents under the Hybrid Alternative. Using this procedure, the highest estimated public MEI risk due to accidents during chemical separation at the Savannah River Site is 0.000033 LCF. The highest estimated accident population risk at the Savannah River Site is 0.24 LCF.

The maximum of the three onsite activities' estimated accident public MEI risks is equal to 0.000033 LCF, which would occur at the Savannah River Site. (The three onsite activities are receipt and management of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory, receipt and temporary management of aluminum-based spent nuclear fuel at the Savannah River Site, and chemical separation at the Savannah River Site). Thus, the chance of this individual incurring an LCF due to this Hybrid Alternative would be less than one in ten thousand.

The total of the three onsite activities' estimated accident population risks is 0.34 LCF.

4.5.5 Summary of the Impacts of the Hybrid Alternative

Principal impacts of the Hybrid Alternative would be occupational and public health and safety impacts. These are presented in Table 4-61 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that one individual with the maximum exposure in each situation would incur an LCF. The population risk expresses the estimated number of additional LCF among the entire exposed population. In general, however, the implementation of the Hybrid Alternative would not pose higher risks than those determined for Management Alternative 1, assuming identical United States site management technology implementation. This is because the analyses in Management Alternative 1 aud its implementation alternatives considered the management of the maximum amount of foreign research reactor spent fuel in the United States.

Table 4-61 Maximum Estimated Radiological Health Impacts of the Hybrid Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	*Population*	
		General Public	*Workers*
Marine Transport			
Incident-Free	0.00052	0	0.024
Accidents	3×10^{-10}	much less than 0.00002	---
Port Activities			
Incident-Free	0.00052	0	0.0076
Accidents	1×10^{-10}	0.00002	---
Ground Transport			
Incident-Free	0.00052	0.11	0.037
Accidents	7.1×10^{-12}	0.000081	---
Site Activities			
Incident-Free	0.026	0.13	0.16
Accidents	0.000033	0.34	---
Highest Individual Risk			
Incident-Free	0.026	----	----
Accidents	0.000033	----	----
Total Population Risk			
Incident-Free	----	0.24	0.23
Accidents	----	0.34	----

Table 4-61 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the DOE management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free individual risk for members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 7.8×10^{-8} LCF.

The highest estimated accident MEI risk is 0.000033 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-61, the total incident-free population risk would be 0.24 LCF for the potentially exposed public, while the corresponding risk would be 0.23 LCF for workers. Thus, there would be an estimated 24 percent chance of incurring one additional LCF among the exposed general public, and a 23 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-61. There is about a seven percent chance that a truck driver or member of the public could die in a traffic accident associated with this Hybrid Alternative. This death would be unrelated to the radioactive nature of the cargo.

4.6 No Action Alternative

Under the No Action Alternative, no foreign research reactor spent nuclear fuel or high-level waste would be accepted into or managed by the United States. The United States would not provide any technical or financial assistance to foreign research reactor operators for the management of their spent nuclear fuel. The United States would rely on the foreign governments' compliance with existing international agreements to control the disposition of foreign research reactor spent nuclear fuel containing uranium enriched in the United States.

Policy Considerations

The No Action Alternative would have a major adverse impact on U.S. nuclear weapons nonproliferation policy. The No Action Alternative would not remove any of the approximately 4.6 metric tons of U.S. origin HEU from international commerce as considered under the proposed action. Under this alternative, the foreign research reactor owners would continue, or may revert back to, use of HEU fuel in their reactors. Countries that can reprocess might send their HEU spent nuclear fuel to be reprocessed and use the separated HEU to produce fresh HEU fuel. In addition, any new research reactors to be built would likely be designed to use HEU fuel. Thus, the No Action Alternative could cause an increase in the number of shipments of weapons-grade nuclear material in transit around the world. It would also damage, perhaps irreparably, the credibility of the RERTR program. Countries that cannot reprocess their research reactor spent nuclear fuel would have to store their fuel. As the spent nuclear fuel ages, it becomes less dangerous to handle (its radioactivity decreases with time), and could possibly become a target of theft and diversion. Hence, the No Action Alternative would undermine the U.S. nuclear weapons nonproliferation policy and the risk of weapons-grade nuclear material being diverted into a nuclear weapons program would increase markedly.

To demonstrate the risk of having reactor owners continue, or revert back to, use of HEU fuel, please see Tables B-3, B-4, and B-5 in Appendix B. These tables list the 104 foreign research reactors whose spent nuclear fuel is included under the proposed action, including 24 reactors that have been converted (fully or partially) or are in various stages of conversion (i.e., ordered, or anticipated to begin converting) from HEU to LEU fuel, and 30 reactors that could be converted, but are not being converted, because the owners of the research reactors are awaiting the outcome of this EIS before they make a decision. Under the No Action Alternative, it is possible that up to 48 foreign research reactor operators could choose to continue or revert back to using HEU fuel in their reactors. These tables also list 23 foreign research reactor operators who possess HEU spent nuclear fuel, even though their reactors are either already shut down or planned to be shutdown for various reasons. This HEU spent nuclear fuel would remain in the foreign research reactor host countries, if the No Action Alternative is selected.

On the other side of the ledger, the benefits obtained from research reactors, described briefly in Section 1.1 of the EIS, would be diminished. Since the No Action Alternative means no U.S. assistance to foreign research reactor operators for managing their spent nuclear fuel, additional research reactors may be forced to shut down, because of lack of funds and/or long term storage capabilities. DOE and the Department of State cannot estimate the number of reactors that would actually be shut down because this would depend on each country's regulations regarding spent nuclear fuel storage. Nevertheless, the medical, industrial and environmental services provided by the shutdown research reactors would be lost. For medical services in particular, foreign research reactors produce radioisotopes used in nuclear medicine in the United States (as discussed in Sections 1.1 and 4.3.1.3 of the EIS). If some of these reactors were forced to shut down, a shortage of medical radioisotopes could occur in the United States. Since the U.S. medical requirements for radioisotopes are not likely to decrease in the near future, alternative sources would have to be found. This could involve an increased level of activity at existing U.S. research reactors or construction of a new reactor in the United States to supply the needed medical radioisotopes, with all the potential environmental impacts of these actions.

Environmental Impacts of Overseas Storage without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved might be greater and the quality of storage technology in some countries might be lower than if the U.S. was involved. Under this option, there would be environmental impacts in foreign countries, but none on U.S. territory, unless some of the material was diverted into nuclear weapons production.

Environmental Impacts of Overseas Reprocessing without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts in foreign countries, but none on U.S. territory, except possibly in cases where some of the material was diverted into nuclear weapons production.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts in foreign countries, but none on U.S. territory. The separated HEU would be more vulnerable to diversion into nuclear weapons production, and the increased reliance on HEU for fuel would increase the number of opportunities for diversion of this weapons grade material.

4.6.1 Overseas Storage Without U.S. Assistance

The material could remain in interim storage overseas. The number of storage sites involved would be greater and the quality of storage technology in some countries might be lower than under the other alternatives. In addition, as the spent nuclear fuel gets older, it becomes less dangerous to handle (its radioactivity decreases with time), and could more easily become a target of theft and diversion.

4.6.2 Overseas Reprocessing Without U.S. Assistance

The material could be reprocessed and the resulting high-level waste could be vitrified or cemented in foreign facilities. Transport of spent nuclear fuel from the reactors to these facilities and the reprocessing activities would produce environmental impacts only in foreign nations.

Under this option, the United States would not accept any shipments of vitrified high-level waste. The transport of vitrified high-level waste back to the country of origin and its storage and disposal would produce environmental impacts only in foreign nations.

4.7 Preferred Alternative

As discussed in detail in Section 2.9, the preferred alternative is to accept and manage the foreign research reactor spent nuclear fuel and target material in the United States. Under this alternative, the aluminum-based foreign research reactor spent nuclear fuel and target material would be transported to and managed at the Savannah River Site. The TRIGA foreign research reactor spent nuclear fuel would be transported to and managed at the Idaho National Engineering Laboratory.

The policy considerations, marine transport impacts, port activities impacts, ground transport impacts, and management site impacts of the preferred alternative presented in this section are based on analysis performed for the basic implementation of Management Alternative 1 (Section 4.2), Implementation Alternative 1c (Section 4.3.1.3), Implementation Alternative 6 (Section 4.3.6), and Implementation Alternative 7 (Section 4.3.7).

4.7.1 Policy Considerations

The policy considerations for the preferred alternative are similar to those described in Section 4.2 for Management Alternative 1. A critical result of implementing this preferred alternative would be the continued viability and vitality of the Reduced Enrichment for Research and Test Reactors (RERTR) Program, which has the goal of minimizing and eventually eliminating the use of HEU in civil nuclear programs by providing foreign research reactor operators with a continued incentive to participate. Similarly, the successful development of alternative fuels for research reactors and the expansion of the program to Russia, the other Newly Independent States, China, South Africa, and other countries, and the establishment of a world-wide norm discouraging the use of HEU, is dependent on a United States' commitment to action such as that embodied in this preferred alternative.

Another crucial consideration associated with the preferred alternative is the *Treaty on the Non-Proliferation of Nuclear Weapons*. The parties to the Non-Proliferation Treaty met in May of 1995 and agreed to extend the treaty indefinitely and without conditions. One key to the success of the 1995 Non-Proliferation Treaty Conference was the ability of the United States to convince other Non-Proliferation Treaty parties that the nuclear weapons states had complied with their obligations under Article IV of the Non-Proliferation Treaty to assist the non-nuclear weapons states with peaceful applications of nuclear energy.

Although the Non-Proliferation Treaty was extended indefinitely, the parties also agreed to review the treaty every five years to ensure that all parties are in compliance. Any country which has been compelled to shut down its research reactors could accuse the United States of not having complied with its treaty obligations. This accusation, however ill-founded, could be made not only by the affected countries, but by any country opposed to the interests of the United States.

Including target material as part of the preferred alternative maximizes the amount of HEU to be removed from international commerce. This includes all the HEU in the basic implementation of Management Alternative 1 [4.6 metric tons (5.1 tons) of heavy metal containing HEU] and all the HEU in the target material in Implementation Subalternative 1c [0.2 metric tons (0.2 tons) of heavy metal containing HEU]. The total amount that would be removed from international commerce is up to 4.8 metric tons (5.3 tons) of heavy metal containing HEU.

DOE's preferred alternative allows for the use of chemical separation under certain circumstances, such as when alternative technologies present higher safety risks, are more costly or are unavailable. If chemical separation is used to process the foreign research reactor spent nuclear fuel, the HEU would be blended down during the separation process to a low-enriched form that is unsuitable for nuclear weapons purposes (the blenddown is also required because the F-Canyon cannot safely process HEU beyond initial dissolution). No plutonium would be separated. Instead, the plutonium would be left in the waste stream with the high-level radioactive chemical separation wastes. In addition, the waste would be handled using technologies that are intended to be used for substantially larger quantities of preexisting wastes (e.g., vitrification of high-level liquid radioactive wastes, grouting for low-level wastes, and incineration for some supernatant).

This potential method of handling the foreign research reactor spent nuclear fuel would be consistent with United States nonproliferation policy, despite the use of chemical separation, because (1) it would reduce the worldwide stockpiles of this nuclear weapons material; (2) no plutonium would be separated; and (3) the chemical separation would not be taking place for either nuclear weapons or nuclear power purposes.

DOE is aware that the inclusion of chemical separation within the preferred alternative could be interpreted by some nations, organizations, and persons as a signal of endorsement of the use of chemical separation as a routine method of waste management for spent nuclear fuel or a reversal of United States policy on chemical separation. This would not be an accurate interpretation. The United States policy regarding chemical separation was established in Presidential Decision Directive 13, and DOE and the Department of State have determined that this preferred alternative is consistent with that policy. The draft version of this EIS indicated that chemical separation is a non-preferred technology. This final preferred alternative includes provision for possible chemical separation. DOE maintains a presumption that spent nuclear fuel would not be chemically separated unless there is an imminent health and safety risk, or other programmatic conditions, that cannot be addressed during the time period when no feasible alternative to chemical separation is available. These considerations will be addressed by the independent study described in Section 2.9.

4.7.2 Marine Transport Impacts

The marine transport impacts of the preferred alternative would be similar to those of the basic implementation of Management Alternative 1, with the addition of the target material shipments. As discussed in Section 4.3.1.3 and Appendix B, Section B.1.5, target material would be prepared for transport by changing it into either oxide or calcine form, and both forms might be accepted at some time during the proposed policy period. Even though it requires less marine transport, the oxide form presents a higher radiological risk under accident conditions because its smaller particle size is more easily dispersed in air. Therefore, to be conservative, the analysis of marine and port radiological accidents is based on the assumption that all the target material would be shipped as an oxide. The rest of the marine and port target material transport analysis is based on the assumption of 15 cask shipments, which is the maximum number of marine target material casks. This represents an increase of approximately two percent over the 721 marine cask shipments in the basic implementation of Management Alternative 1.

Marine transport to the West Coast of the United States would be limited to a maximum of approximately 38 casks, which slightly decreases the total number of days the ships would be at sea. Furthermore, DOE would strive to minimize the number of shipments necessary by coordinating shipments from several reactors at a time (i.e., by placing multiple casks [up to 8] on a ship). DOE currently estimates that approximately 5 shipments through the Naval Weapons Station at Concord, California would be necessary.

Impacts of Incident-Free Marine Transport

The highest estimated maximally exposed worker risk due to foreign research reactor spent nuclear fuel is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years (Table 4-2). This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for all of the ships' crews involved in the marine transport of foreign research reactor spent nuclear fuel is about 0.034 LCF, as discussed in Section 4.2.1.2.

Target material contains far less radioactivity than foreign research reactor spent nuclear fuel. Each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 marine casks of target material.

Impacts of Accidents During Marine Transport

The risks associated with accidents at sea are bounded by the risks of the same accidents in ports because humans in the vicinity of accidents at sea are much fewer in number than even the least populated port.

Marine Transport Cumulative Impacts and Mitigation Measures

The marine transport cumulative impacts and mitigation measures for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.1.4 and 4.2.1.5, respectively.

4.7.3 Port Activities Impacts

Although all of the candidate ports of entry presented in Section 3 are acceptable, based on the port selection criteria described in Appendix D, DOE would prefer to use military ports. All aluminum-based foreign research reactor spent nuclear fuel and target material from overseas would arrive at candidate ports on the East Coast of the United States, preferably the Naval Weapons Station at Charleston, South Carolina. Up to approximately 38 casks of TRIGA foreign research reactor spent nuclear fuel would arrive at candidate ports on the West Coast of the United States, preferably the Naval Weapons Station at Concord, California.

Impacts of Incident-Free Port Activities

As shown in Table 4-5, the highest maximally exposed worker risk is 0.00052 LCF, which is based on the conservative assumption that one individual receives the maximum annual dose (100 mrem) every year for 13 years. This means that the chance of this hypothetical individual incurring a latent cancer due to the preferred alternative would be less than one in a thousand.

The highest estimated population risk for port workers is about 0.012 LCF, as discussed in Section 4.2.2.3.

As discussed under *Impacts of Incident-Free Marine Transport* above, each transportation cask of target material would produce a radiation dose rate far below the rate that was assumed for the foreign research reactor spent nuclear fuel. Thus, the rounded-off results of the incident-free radiological risk calculations for the basic implementation of Management Alternative 1 are not affected by the addition of up to 15 cask shipments of target material.

Impacts of Accidents During Port Activities

The radiological risks due to port accidents were estimated in the same manner as for the basic implementation (Section 4.2.2.3) and Implementation Alternative 1c (Section 4.3.1.3) of Management Alternative 1. The highest estimated population risk for the entire preferred alternative program is 7.1×10^{-7} LCF. This risk estimate is lower than the earlier alternatives due to the use of military ports in the preferred alternative. These ports are located in areas of low population density, so the number of people potentially affected is much lower. The addition of target material causes a very small incremental increase (3×10^{-9} LCF) in the risk.

Port Activities Cumulative Impacts, Mitigation Measures, and Environmental Justice

The port activities cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.2.4, 4.2.2.5, and 4.2.2.6, respectively.

4.7.4 Ground Transport Impacts

The ground transport impacts were calculated under the assumption that only military ports would be used. DOE has selected military ports close to the management sites (the Charleston NWS in South Carolina and the Concord NWS in California) as the preferred ports of entry.

The risk estimates were maximized by assuming all target material would be oxide for radiological accident calculations and calcine for all other calculations. The calcine form could require up to 125 casks of target material to be shipped overland from Canada.

The preferred points of entry, destinations, and approximate numbers of cask shipments in the preferred alternative are presented in Table 4-62. Other shipment distributions would also be possible.

Table 4-62 Points of Entry, Destinations, and Numbers of Shipments in the Preferred Alternative

Cargo and Destination	*Point of Entry*			*Total Cask Shipments*
	East Coast	*West Coast*	*Canadian Border*	
Aluminum-Based Foreign Research Reactor Spent Nuclear Fuel to the Savannah River Site	559	0	116	675
TRIGA Foreign Research Reactor Spent Nuclear Fuel to the Idaho National Engineering Laboratory	124	38	0	162
Target Material to the Savannah River Site	up to 15	0	up to 125	up to 140
Total Cask Shipments	**up to 698**	**38**	**up to 241**	**up to 977**

Impacts of Incident-Free Ground Transport

The incident-free ground transport of foreign research reactor spent nuclear fuel and target material is estimated to result in a maximum of 0.089 LCF over the entire duration of the program. This is the sum of the estimated number of radiation-related LCF to the public and transportation workers.

The estimated maximum number of radiation-related LCF for transportation workers is 0.022. The estimated maximum number of radiation-related LCF for the general public is 0.067, and the estimated maximum number of non-radiation-related fatalities from vehicular emissions is 0.018.

Impacts of Accidents During Ground Transport

The total ground transport accident population risks for the preferred alternative are estimated to be less than 0.00072 LCF from radiation and 0.052 from traffic collisions.

The maximum foreseeable offsite transportation accident would involve a transportation cask of oxide target material in a suburban population zone under neutral (average) weather conditions, which could expose the MEI to 150 mrem. A similar event involving a transportation cask of spent nuclear fuel could expose the MEI to 2.4 mrem. These events are both in the highest accident severity category. Taking all the possible consequences and frequencies of these accidents into account, and adding the foreign research reactor spent nuclear fuel risks with the target material risks yields the MEI risk of 2.7×10^{-11} LCF for the preferred alternative.

Ground Transport Cumulative Impacts, Mitigation Measures, and Environmental Justice

The ground transport cumulative impacts, mitigation measures, and environmental justice for the preferred alternative would be the same as for the basic implementation of Management Alternative 1, which are discussed in Sections 4.2.3.4, 4.2.3.5, and 4.2.3.7, respectively.

4.7.5 Management Site Impacts

As discussed in Section 2.9, all the TRIGA foreign research reactor spent nuclear fuel would be managed at the Idaho National Engineering Laboratory. The fuel would be received and stored in existing facilities. The environmental impacts of the preferred alternative at the Idaho National Engineering Laboratory can be estimated from the environmental impact analysis presented for the basic implementation of Management Alternative 1 (Section 4.2).

At the Savannah River Site, however, the impacts would vary depending on the specific outcome of the preferred management strategy at the site. Aluminum-based foreign research reactor spent nuclear fuel and target material would be managed at the Savannah River Site. The management of the foreign research reactor spent nuclear fuel is based on the schedule for successful implementation of a new treatment and/or packaging technology. If such a new technology could not be successfully demonstrated by the year 2000, chemical separation of a portion of the foreign research reactor spent nuclear fuel might be implemented. The foreign research reactor spent nuclear fuel and target material that is not chemically separated would be stored in existing facilities at the Savannah River Site until the new technology is operational.

Since the preferred alternative includes the construction and operation of an unspecified treatment and/or packaging technology at the Savannah River Site, the environmental impacts of this alternative at this site cannot be estimated with precision. DOE expects, however, that the radiological and nonradiological health and environmental effects from the construction and operation of facilities that would support a new technology would not exceed those estimated for construction of new dry storage facilities and operation of a conventional chemical separation facility evaluated in Sections 4.2.4.2 and 4.3.6 of this EIS. This expectation is based on the following general principles:

- New facilities would be constructed using current DOE design criteria which have evolved on the basis of increased protection of the public, workers, and the environment.
- The primary source of radiological releases from the chemical separation process is the front end dissolution of the spent nuclear fuel matrix. None of the new technologies considered involves a process that would produce greater releases.
- One of the reasons for the development of a new treatment and/or packaging technology is to reduce the volume and toxic nature of low-level and hazardous waste streams, an issue considered to be a disadvantage of the chemical separation process.

Nonradiological impacts from the construction of facilities that would support the new technology are expected to be typical to those assessed for the construction of new staging and storage facilities assessed for the basic implementation of Management Alternative 1 in Section 4.2.4.2. These include land use, socioeconomics, cultural resources, aesthetic and scenic resources, geology, air and water quality, ecology, noise, materials and energy consumption, and non-radiological or non-toxic waste production during construction.

The occupational and public health and safety, waste management, and cumulative impacts presented below assume that the implementation of the preferred alternative at the Savannah River Site would result in radiological health effects equal to those presented in Sections 4.3.6 and 4.3.7 of this EIS.

4.7.5.1 Occupational and Public Health and Safety

Impacts to the Public of Incident-Free Management Site Activities

The approximately 4,900 foreign research reactor spent nuclear fuel elements that would be received and managed at the Idaho National Engineering Laboratory under the preferred alternative represent about 22 percent of the total number of elements that would be received and managed there under the basic implementation of Management Alternative 1. Annual public impacts due to incident-free emissions from both aluminum-based and TRIGA foreign research reactor spent nuclear fuel during receipt and management at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 are presented in Table 4-9. Applying these results to the preferred alternative at the Idaho National Engineering Laboratory for only TRIGA spent nuclear fuel is conservative because the TRIGA spent nuclear fuel would produce less gaseous fission product emissions than the mixture of spent nuclear fuel in the basic implementation of Management Alternative 1. Adjusting the figures from Table 4-9 to account for the reduced amount of material in the preferred alternative yields the highest estimated risks for this part of the preferred alternative. The highest estimated public MEI risk is 7.8×10^{-10} LCF and the highest estimated public population risk is 0.0000064 LCF.

Radioactive emissions would not be expected from the target material receipt or storage because this material contains no gaseous fission products. Therefore, the incident-free radiological impacts to the public would be zero.

The incident-free radiological public health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site.* The highest estimated public MEI risk is 0.0000043 LCF and the highest estimated public population risk is 0.18 LCF.

The maximum of the onsite activities' estimated public incident-free MEI risks is equal to 0.0000043 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in one hundred thousand.

The total of the onsite activities' estimated incident-free population risks to the people who live near both sites is equal to 0.18 LCF. This number means that there would be an approximately 18 percent chance of one additional LCF among the population residing around the two sites due to these incident-free activities.

Impacts to Workers of Incident-Free Management Site Activities

Incident-free maximally exposed worker radiation dose depends upon the duration of the receipts, not the amount of spent nuclear fuel involved. The duration of the receipts in the preferred alternative is 13 years, the same as that in the basic implementation, the target material alternative, and the chemical separation alternative of Management Alternative 1. Thus, the estimated maximally exposed worker dose is also the same. The highest maximally exposed worker risk is estimated to be 0.026 LCF.

The incident-free worker population risks of the basic implementation of Management Alternative 1 at the Idaho National Engineering Laboratory were presented in Section 4.2.4.1. Using the same evaluation process yields a dose of 52 person-rem (dry storage in existing facilities). The associated worker population risk for this part of the preferred alternative is 0.021 LCF.

The incident-free radiological worker health impacts at the Savannah River Site due to the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Incident-Free Impacts at the Savannah River Site*. The highest estimated worker population risk is 0.21 LCF.

The total of the onsite activities' estimated incident-free worker population risks at both sites is 0.23 LCF, which means that there would be an approximately 23 percent chance of one additional LCF among the affected radiation workers at the two sites.

Impacts to the Public of Accidents Onsite

Accident scenarios, frequencies, consequences, and risks for the preferred alternative at the Idaho National Engineering Laboratory are the same as those for the basic implementation of Management Alternative 1. The estimated accident frequencies and consequences are presented in Table 4-20. The highest estimated public MEI/NPAI consequence is 0.000015 LCF and the highest estimated public population consequence is 1.0 LCF. Annual accident risks for receipt, unloading, and storage at the Idaho National Engineering Laboratory are presented in Table 4-25. Multiplying these figures by the appropriate duration of the activity (13 years for receipt and 40 years for storage) yields the risk due to accidents at the Idaho National Engineering Laboratory. These estimates are conservative because the TRIGA spent nuclear fuel involved would release fewer fission products than would the bounding radionuclide inventory presented in Appendix B, Table B-6 that was used for the evaluations in the basic implementation of Management Alternative 1. The highest estimated accident MEI/NPAI risk for this part of the preferred alternative is 0.0000019 LCF, which is due to a criticality event at an existing wet storage facility. The highest estimated accident population risk for this part of the preferred alternative is 0.016 LCF, which is due to an accidental fuel assembly breach.

The radiological public health impacts due to accidents at the Savannah River Site under the preferred alternative are assumed to be equal to those discussed in Section 4.3.6.6.4 under the subheading, *Impacts of Chemical Separations Accidents at the Savannah River Site*. The highest estimated public MEI risk is 0.000047 LCF and the highest estimated public population risk is 0.43 LCF.

The maximum of the onsite activities' estimated accident public MEI risks is equal to 0.000047 LCF, which would occur at the Savannah River Site. The chance of this hypothetical individual incurring an LCF due to the preferred alternative would be less than one in ten thousand.

The total of the onsite activities' estimated accident population risks at both sites is equal to 0.45 LCF. This means that there would be an approximately 45 percent chance that one additional LCF would be incurred among the people living near both sites due to accidents during these activities.

4.7.5.2 Waste Management

Implementation of the receipt and storage portions of the preferred alternative would introduce a very small increase in waste generation over current levels at both sites. Baseline site generation of waste is shown in Appendix F, Tables F-23 and F-46 for the Savannah River Site and the Idaho National Engineering Laboratory, respectively. It should be noted that the figures represent storage of more fuel elements, at both sites, than the amounts indicated by the preferred alternative. Implementation of a new technology would produce waste in the amounts presented in Table 4-56.

If the chemical separation portion of the preferred alternative is implemented, this would generate different wastes at the Savannah River Site in place of some of the waste from the new technology. As discussed in Section 4.3.6.6.5, the primary wastes generated during conventional chemical separation and vitrification operations are high-level waste glass in canisters and saltstone. Assuming the chemical separation portion of the preferred alternative could involve up to approximately one-third of the aluminum-based foreign research reactor spent nuclear fuel (6,000 elements), this waste generation would be about one-third of the amount generated under Implementation Alternative 6. Under the preferred alternative, DOE could generate up to approximately 24 high-level waste glass canisters and 1,350 cubic meters (47,700 cubic feet) of saltstone. These wastes would be managed along with much larger quantities of identical wastes in existing facilities at the Savannah River Site.

4.7.5.3 Cumulative Impacts

Cumulative impacts from the implementation of the preferred alternative at both the Idaho National Engineering Laboratory and the Savannah River Site are expected to be lower than those presented for the basic implementation of Management Alternative 1 in Sections 4.2.4.3.1 and 4.2.4.3.2 for the two sites, respectively. At both sites the cumulative impacts from the management of foreign research reactor spent nuclear fuel and impacts from other existing or planned activities or actions at the sites, as presented in Tables 4-29 and 4-30 for Savannah River Site and Idaho National Engineering Laboratory, respectively, including activities not related to the management of spent nuclear fuel, would not challenge or have detrimental effects on the public or environmental resources at the sites.

4.7.5.4 Mitigation Measures

Although environmental impacts at both the Savannah River Site and the Idaho National Engineering Laboratory for the implementation of the preferred alternative would be minimal in all environmental media and mitigation measures would not be necessary, the sites would implement measures in some areas to minimize impacts. Such measures would be taken in the areas of pollution control, socioeconomics, cultural resources, air and water resources, occupational and public health and safety, and accident prevention. Section 4.2.4.6 provides details on these issues.

4.7.5.5 Environmental Justice

The environmental justice conclusions for the management sites discussed in Section 4.2.4.5 for the implementation of Management Alternative 1 are valid for the preferred alternative. As discussed in Section 4.2.4.5, minority or low-income populations living near the Savannah River Site or the Idaho National Engineering Laboratory would not be subjected to any disproportionately high and adverse impacts.

4.7.6 Short Term Uses and Long Term Productivity

The use of land at the Savannah River Site for the potential construction of the new technology facilities would conform with the land use policy at the site. After adoption of an overall strategy for the management of all DOE-owned spent nuclear fuel (including spent nuclear fuel from foreign research reactors), some of the areas may be released for other productive uses.

4.7.7 Irreversible and Irretrievable Commitments of Resources

The operation of existing storage facilities at both sites would involve the consumption of some irretrievable amounts of electrical energy. The potential construction of new technology facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in the construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

4.7.8 Summary of the Impacts of the Preferred Alternative

The principal impacts of the preferred alternative would be occupational and public health and safety impacts. These are presented in Table 4-63 in terms of the risk of death due to cancer during each of the four segments of this alternative. The table also shows, in the bottom rows, the highest of the individual risks and the total of the population risks. Each individual risk expresses the probability that the one individual with the maximum exposure in each situation would incur an LCF due to the preferred alternative. The population risk expresses the estimated number of additional LCF among the entire potentially exposed population.

Table 4-63 shows that the greatest radiological risks would occur during ground transport or management site activities. These results are based on conservative assumptions, including: (1) every package of foreign research reactor spent nuclear fuel producing a dose rate equal to the regulatory limit; (2) every truck shipment exposing people at highway rest stops for times about equal to the actual driving times; and (3) one individual at the management site receiving the maximum dose allowed by DOE regulation every year.

The highest estimated incident-free individual risk is 0.026 LCF, which would apply to an onsite radiation worker. This individual would have a 2.6 percent chance of incurring an LCF. DOE and the Department of State believe the actual risk would be much lower due to administrative procedures such as worker rotation. The highest estimated incident-free risk for individual members of the public is much lower than the maximally exposed worker risk. DOE estimates this risk to be approximately 0.0000043 LCF.

The highest estimated accident MEI risk is 0.000047 LCF, which applies to a hypothetical member of the public who lives at the site boundary. This individual's chance of incurring an LCF due to an accident under this alternative would be less than one in ten thousand. The accident risk to workers is discussed qualitatively in Section 4.2.4.1 under the heading, "Impacts of Accidents to Close-in Workers."

As shown in Table 4-63, the total incident-free population risk would be 0.25 LCF for the potentially exposed public, while the corresponding risk would be 0.30 LCF for workers. Thus, there would be an estimated 25 percent chance of incurring one additional LCF among the exposed general public, and a 30 percent chance of incurring one additional LCF among workers. The chance of incurring two additional LCFs among each population group would be even lower.

Table 4-63 Maximum Estimated Radiological Health Impacts of the Preferred Alternative

	Risks (LCF)		
	Maximally Exposed Worker, MEI, or NPAI	Population	
		General Public	Workers
Marine Transport			
Incident-Free	0.00052	0	0.034
Accidents	5×10^{-10}	much less than 7.1×10^{-7}	---
Port Activities			
Incident-Free	0.00052	0	0.012
Accidents	2.9×10^{-10}	7.1×10^{-7}	---
Ground Transport			
Incident-Free	0.00052	0.067	0.022
Accidents	2.7×10^{-11}	0.00072	---
Site Activities			
Incident-Free	0.026	0.18	0.23
Accidents	0.000047	0.45	---
Highest Individual Risk			
Incident-Free	0.026	----	---
Accidents	0.000047	----	----
Total Population Risk			
Incident-Free	----	0.25	0.30
Accidents	----	0.45	----

Deaths due to traffic accident trauma and LCF due to vehicle emissions are not included in Table 4-63. There is approximately a five percent chance that a truck driver or member of the public could die in a traffic accident associated with the preferred alternative. This death would be unrelated to the radioactive nature of the cargo.

4.8 Comparison of the Alternatives

This chapter has identified the policy considerations and potential environmental impacts resulting from the proposed action, with all of its various alternatives, and the No Action Alternative. This section provides a comparison of the potential impacts of each alternative, with emphasis on key issues such as the amount of HEU removed from international commerce and risks to workers and the public.

4.8.1 Amount of HEU Removed from International Commerce

The purpose and need for Agency action is driven by the concern that HEU in civilian commerce might be diverted into a nuclear weapons program. Removal of HEU from international civilian commerce will greatly enhance the goals of the U.S. nuclear weapons nonproliferation policy. Figure 4-21 compares the quantities of HEU that would be removed from international civil commerce under the basic implementation of Management Alternative 1, the implementation alternatives, the Hybrid Alternative, the No Action Alternative, and the preferred alternative.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would remove up to an estimated 4.6 metric tons (5.1 tons) of HEU from international commerce. By accepting this weapons-grade material into the United States for storage, the risk of material diversion would be eliminated. For comparison, the United States moved about 0.6 metric tons (0.7 tons) of HEU from Kazakhstan to the United States in November and December 1994 to ensure that it could not be diverted into a nuclear weapons program. The quantity of HEU involved in the basic implementation of Management Alternative 1 is over seven times the amount removed from Kazakhstan. The HEU in foreign research reactor spent nuclear fuel, however, is mixed with fission products, so it would require more sophisticated chemical processing to convert it to uranium metal suitable for use in nuclear weapons.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amounts identified in the basic implementation of Management Alternative 1 could have an impact on the amount of HEU in international civil commerce. As shown in Figure 4-21, the implementation alternative of accepting spent nuclear fuel only from developing nations would remove up to only about 0.24 metric tons (0.26 tons) of HEU from international commerce. The implementation alternative of accepting target material in addition to the foreign research reactor spent nuclear fuel in the basic implementation of Management Alternative 1 would remove the most HEU (up to 4.8 metric tons or 5.3 tons) from international commerce. If the acceptance policy lasted for only 5 years, then the amount of HEU involved would be only up to 4.1 metric tons (4.5 tons).

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 could indirectly impact the amount of HEU removed from international commerce depending on whether those financial adjustments influence the amount of foreign research

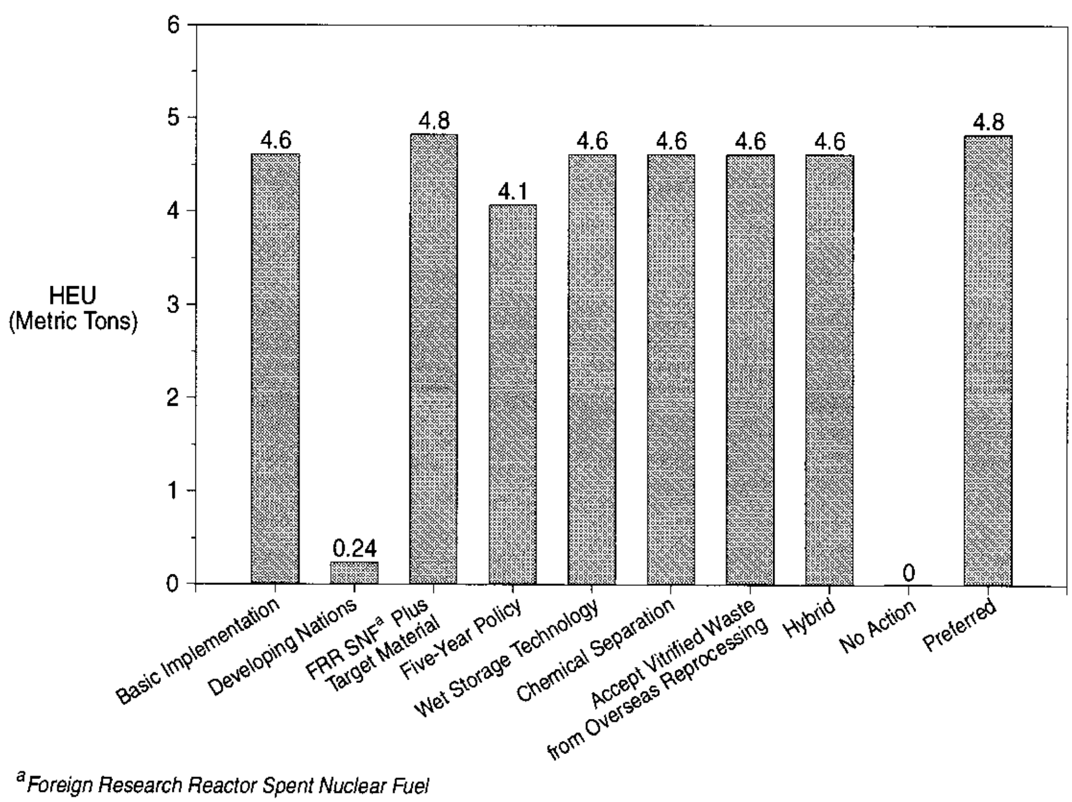


Figure 4-21 Quantities of HEU that Would Be Removed from International Commerce Under Each Alternative

reactor spent nuclear fuel transported to the United States. The final amount of HEU removed from international civil commerce through the application of different financial arrangements cannot be readily determined at this point.

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international commerce, i.e., the action would still remove up to 4.6 metric tons (5.1 tons) of HEU. Similarly, the use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would not change the amount of HEU removed from international civil commerce, since the alternative relates to actions within the United States. Implementation by use of near term chemical separation in the United States instead of interim storage would also cause no change in the amount of HEU removed, again because the alternative involves actions in the United States.

Storing foreign research reactor spent nuclear fuel at one or more overseas sites would have a questionable effect on the amount of HEU removed from international commerce. Although this management alternative would provide the United States some limited measure of control over the foreign research reactor spent nuclear fuel, the prevention of material diversion into a nuclear weapons program would not be as fully ensured as if the foreign research reactor spent nuclear fuel was accepted into the United States. This alternative would leave HEU stockpiled around the world.

The implementation alternative of overseas reprocessing would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

Hybrid Alternative: The Hybrid Alternative chosen for analysis would remove the same amount of HEU from international commerce as would the basic implementation of Management Alternative 1, independent of decisions on the management of the resulting high-level waste.

No Action Alternative: Under this alternative, the United States would rely solely on the foreign governments' compliance with international agreements to control the foreign research reactor spent nuclear fuel. A policy of no action by DOE and the Department of State runs counter to U.S. nuclear weapons nonproliferation policy by causing continued reliance on HEU, thus not realizing the goal of eliminating civil commerce in HEU.

Preferred Alternative: The preferred alternative would remove the same amount of HEU (up to 4.8 metric tons or 5.3 tons) from international commerce as would Implementation Alternative 1c of Management Alternative 1. This amount is higher than for the other alternatives.

4.8.2 Radiological Risk to Individuals

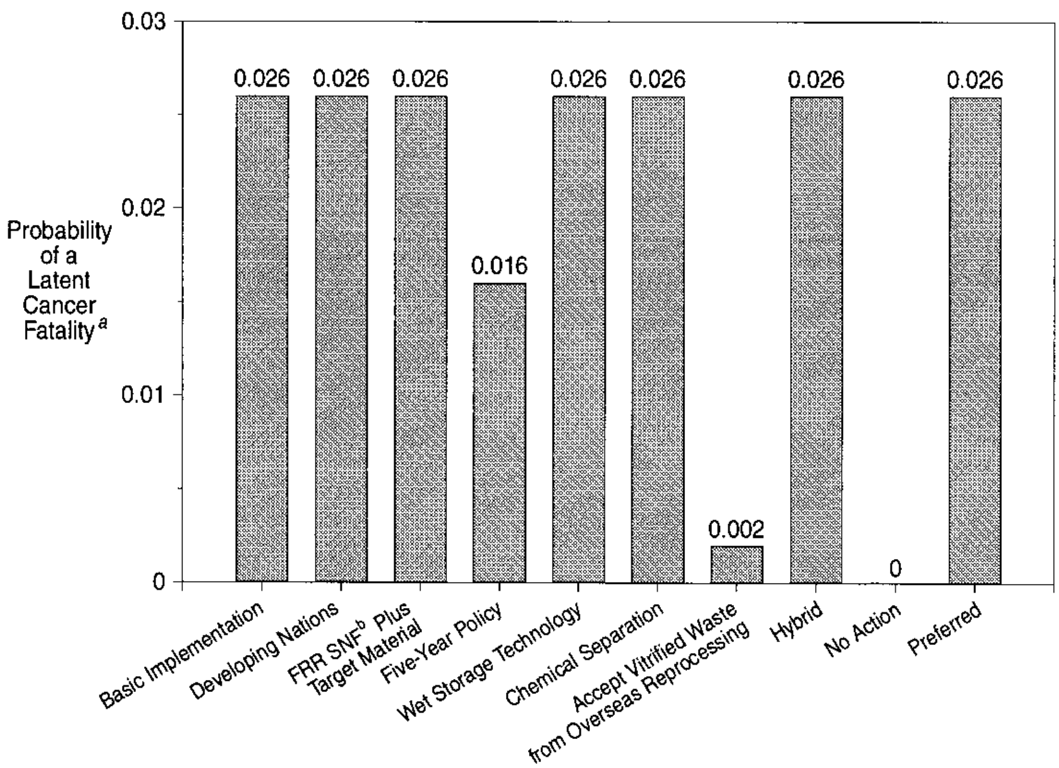
A maximally exposed worker or an MEI in the public is a hypothetical individual who records the highest possible exposure to radiation in a given situation, and the associated risks are different depending on the alternative considered. Figures 4-22 and 4-23 present comparisons of the estimated radiological risk to the maximally exposed worker and to the MEI under each alternative for incident-free and accident conditions, respectively. Alternatives involving the smallest number of cask shipments into the United States would produce the lowest individual risks. There would be no maximally exposed worker risk or MEI risk in the United States under the No Action Alternative.

The incident-free maximally exposed worker risk estimates are driven by the assumption that a radiation worker would receive the maximum radiation dose allowed by law for every year that foreign research reactor spent nuclear fuel is accepted. This risk depends only on the duration of the action, not on the number of casks or elements. Thus, the Five-Year Acceptance Alternative would present lower risk than the alternatives which last for 13 years.

The accident MEI risk estimates are dominated by onsite accident scenarios. This is because during marine transport, port activities, and ground transport, the foreign research reactor spent nuclear fuel would be inside transportation casks. During onsite activities, while spent nuclear fuel is outside of transportation casks, the probability of an incident that could release radioactive material is higher. The highest estimated accident MEI risk in the public is 0.00015 LCF, which means that this hypothetical individual's increased chance of incurring an LCF would be less than two in ten thousand.

4.8.3 Radiological Risk to Exposed Populations

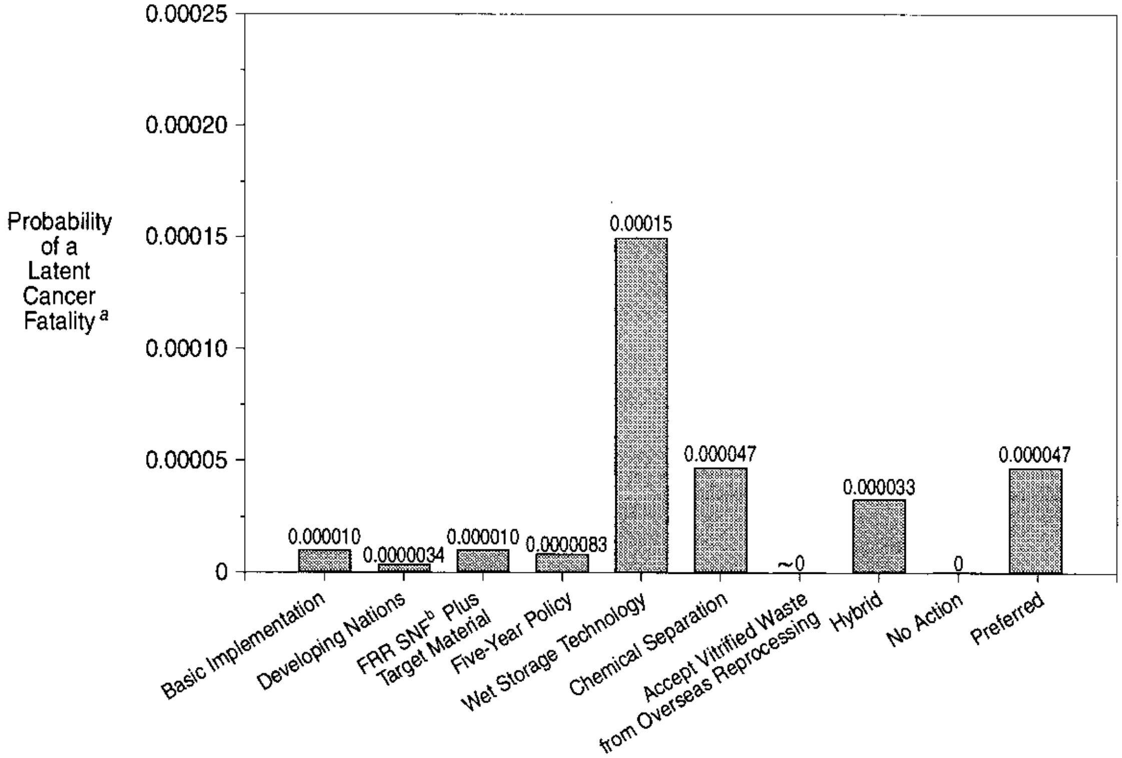
Population risk is the risk of additional latent cancers occurring among people (both public and workers) who would be exposed to radiation. Risks vary with the alternative considered. Figures 4-24 and 4-25 present comparisons of the estimated incident-free radiological risks to the public and worker populations under each alternative. Alternatives involving the smallest number of cask shipments into the United



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-22 Maximum Estimated Incident-Free Radiological Risk to the Maximally Exposed Worker Under Each Alternative



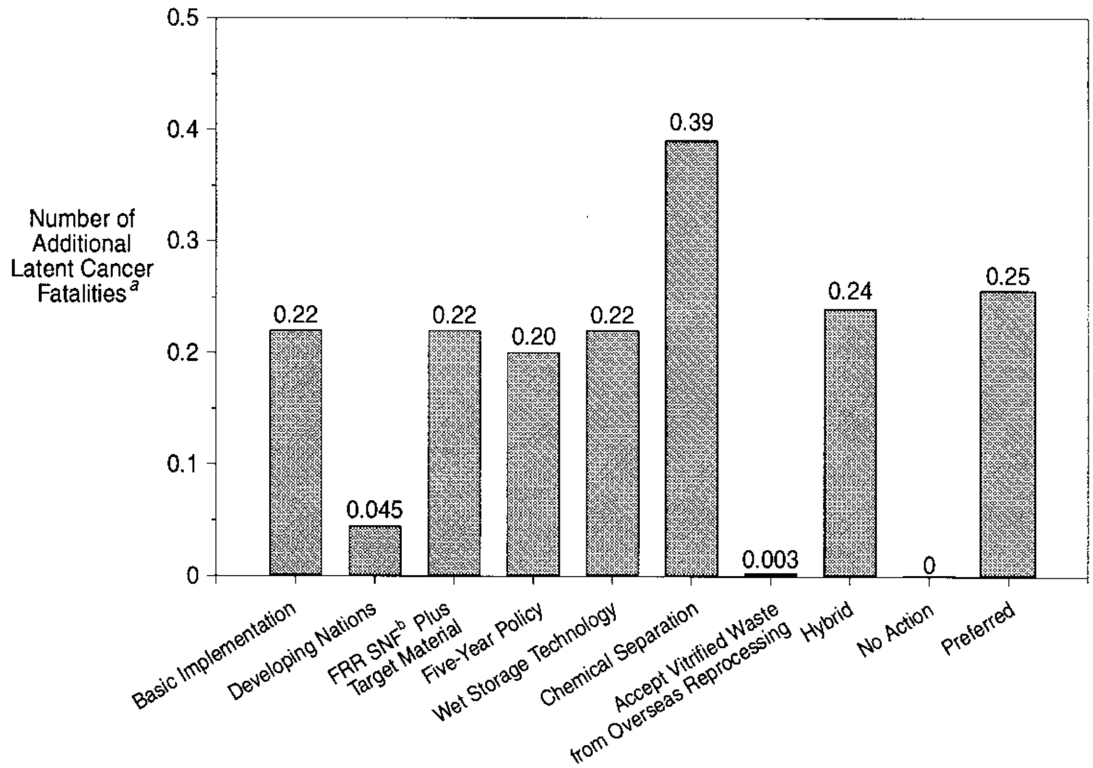
[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-23 Maximum Estimated Accident Radiological Risk to the MEI in the Public Under Each Alternative

States would produce the lowest population risks. The chemical separation, overseas reprocessing, and preferred alternative are the alternatives in which the waste would be conditioned for disposal. Under the other alternatives, some form of processing may be required at some time in the future before disposal. There would be no population risk in the United States under the No Action Alternative. Under all the alternatives the estimated incident-free public and worker population risks would result in less than one-half additional LCF among each population group.

Figure 4-26 presents a comparison of the estimated accident radiological population risks to the public under each alternative. Those alternatives involving some form of processing in the United States would present the largest accident risks, but these risks would occur in the near term. Under the other alternatives, some form of processing may be required at some time in the future before disposal. Under all the alternatives, the estimated accident public population risks would result in less than one-half additional LCF.



[a] Impacts evaluated were those in the United States and on the Global Commons.

[b] Foreign Research Reactor Spent Nuclear Fuel

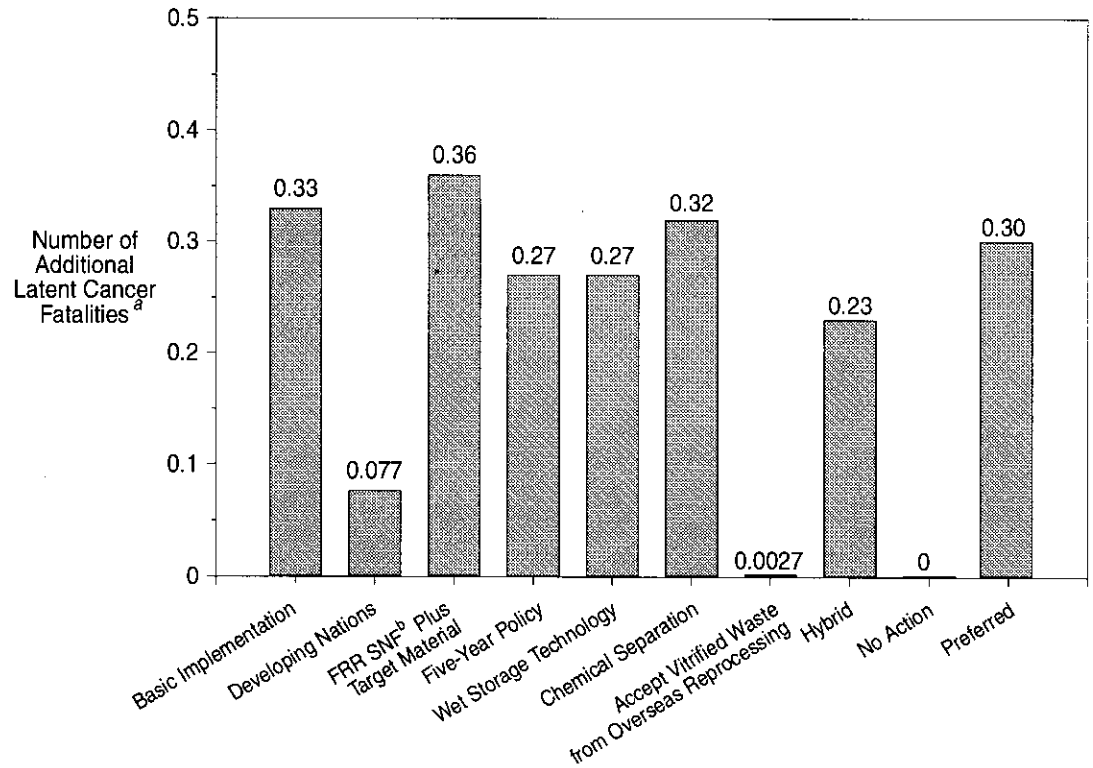
Figure 4-24 Maximum Estimated Incident-Free Radiological Population Risk to the General Public Under Each Alternative

4.8.4 Nonradiological Risks

The transport of foreign research reactor spent nuclear fuel from the ports to the sites would involve some risk of death due to traffic accidents for both the truck drivers and the public. Figure 4-27 presents a comparison of the estimated traffic accident risk to both the drivers and public combined under each alternative. Estimates include the risks associated with transporting the empty casks back to the ports.

Results are directly proportional to the number of highway miles over which casks would be transported under each alternative. The basic implementation of Management Alternative 1 and four of the implementation alternatives would have essentially the same risk, while the Developing Nations Subalternative and the Hybrid Alternative would have lower traffic accident risks.

Under the subalternative of accepting vitrified waste from overseas reprocessing, an estimated eight cask shipments would be accepted in the United States, so the traffic accident risk would be extremely low. There would be no population risk in the United States under the other overseas subalternative, as well as the No Action Alternative.



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-25 Maximum Estimated Incident-Free Radiological Population Risk to Workers Under Each Alternative

The traffic accident risk is also relatively low under the preferred alternative because all the cask shipments of aluminum-based foreign research reactor spent nuclear fuel would go through an east coast port or ports to the Savannah River Site. This effectively minimizes the ground transport risk by minimizing the number of highway miles required.

4.8.5 Land Use

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major land use issues at any of the potential foreign research reactor spent nuclear fuel management sites. If additional storage space were required for the foreign research reactor spent nuclear fuel, the space would be built on DOE-owned lands, inside the boundaries of the DOE management sites.

Implementation Alternatives: Acceptance of amounts of foreign research reactor spent nuclear fuel different from the amount identified in the basic implementation of Management Alternative 1 would not cause land use issues, even though storage needs may vary due to the United States receiving a larger (if target material is accepted in addition to spent nuclear fuel) or smaller (e.g., from developing nations only)

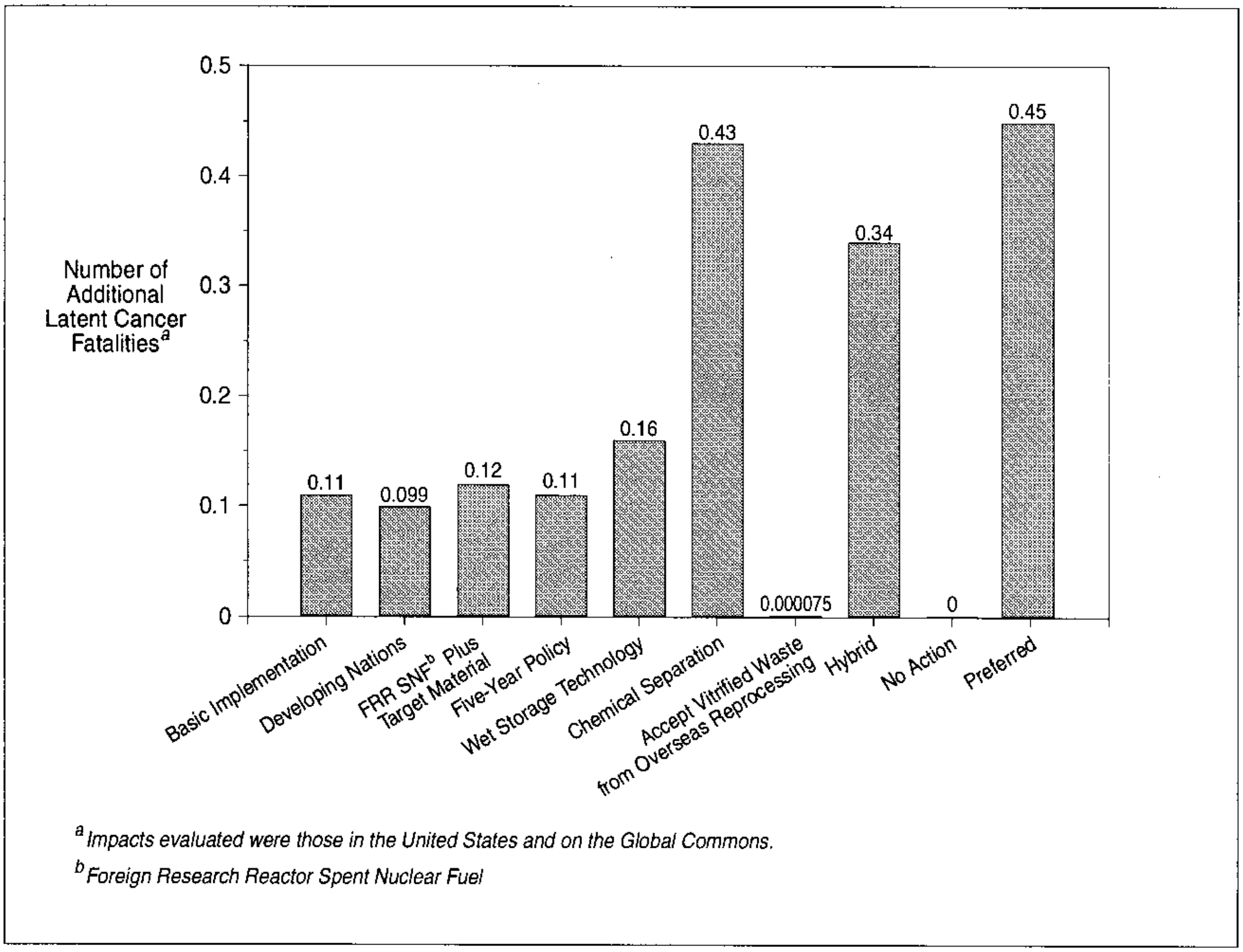
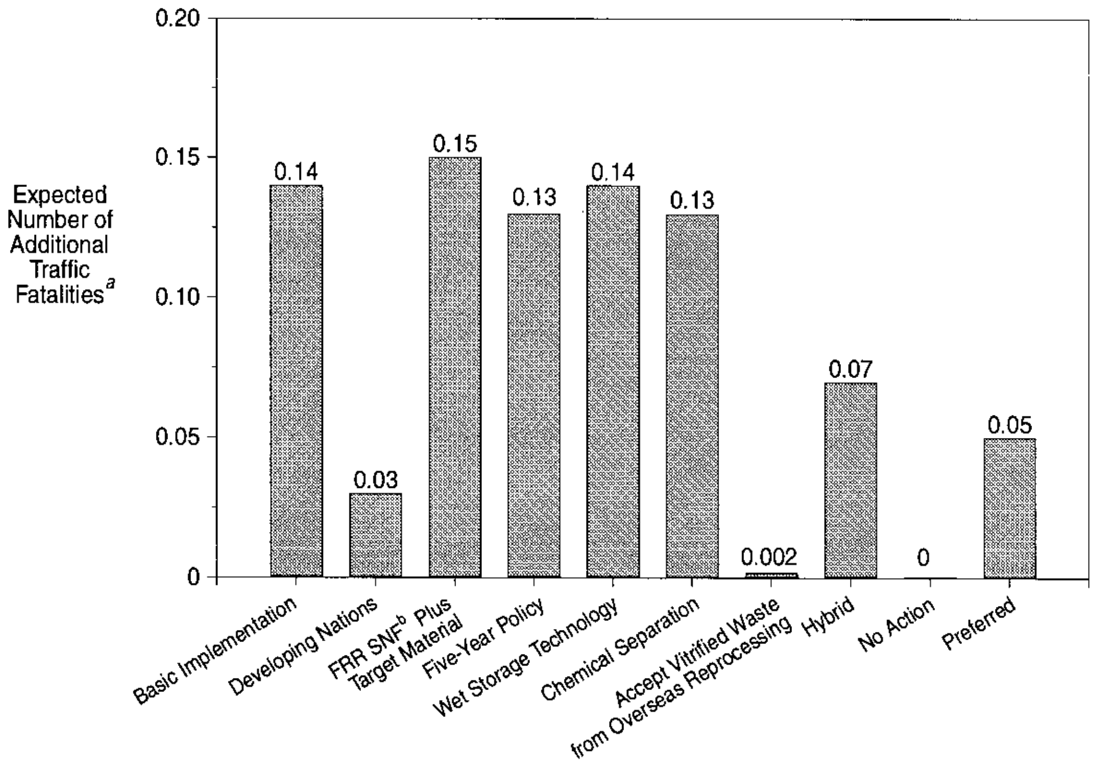


Figure 4-26 Maximum Estimated Accident Radiological Population Risk to the General Public Under Each Alternative

amount of material than identified in the basic implementation of Management Alternative 1. As mentioned above, additional storage space, if required, would be created on DOE-owned land, creating no outside land use issues.

Acceptance of foreign research reactor spent nuclear fuel for periods of time different from the time periods identified in the basic implementation of Management Alternative 1 would not cause any land use issues as the timeframe would not necessarily change the amount of foreign research reactor spent nuclear fuel received by the United States. If a policy of 5 years of acceptance was instituted, less spent nuclear fuel would be received by the United States, and if an indefinite HEU/10-year LEU policy were to be adopted, storage space would be created on DOE management sites, causing no issues in relation to outside lands.

Implementation through financial arrangements different from those identified in the basic implementation of Management Alternative 1 would have no impact on land use, as this alternative would have no effect on lands not owned by DOE.



[a] *Impacts evaluated were those in the United States and on the Global Commons.*

[b] *Foreign Research Reactor Spent Nuclear Fuel*

Figure 4-27 Maximum Estimated Traffic Accident Risk Under Each Alternative

Implementation by taking title to the foreign research reactor spent nuclear fuel at locations different from those identified in the basic implementation of Management Alternative 1 would cause no land use issues, as it would have no effect on the storage needs or the amount of foreign research reactor spent nuclear fuel received by the United States.

Use of wet storage technology for the interim period instead of dry storage technology as identified in the basic implementation of Management Alternative 1 would cause no land use issues, as the storage facilities (wet or dry) would be on DOE-owned land, and would have no effect on outside (non-DOE-owned) lands. If DOE decides to purchase the BNFP facility for interim wet storage, however, this would require adding some land to the Savannah River Site.

Implementation by use of near term chemical separation in the United States instead of interim storage would have no impact on land use, as the separation would be performed on DOE-owned land, with no effect on outside (non-DOE-owned) lands.

Similarly, there would be no land use concerns under either of the overseas subalternatives or the Hybrid Alternative presented in this EIS. A policy of no action (the No Action Alternative) regarding foreign research reactor spent nuclear fuel would cause no land use issues in the United States.

Land use for construction under the preferred alternative would be similar to the land use for construction under the basic implementation of Management Alternative 1.

4.8.6 Cultural Resources

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not result in major impact to the cultural resources of the management sites being considered for the storage of the foreign research reactor spent nuclear fuel. Although the sites have not been evaluated and audited for cultural resources, surveys would be completed prior to any construction or other activity that would potentially disturb these areas. Areas of cultural or historical significance are protected by laws and acts (e.g., Native American Grave and Repatriation Act, National Historic Preservation Act, etc.), and the basic implementation of Management Alternative 1 is not likely to have an impact on areas of cultural or historical significance.

Implementation Alternatives: Since the safety of areas of cultural or historic significance is protected under the basic implementation of Management Alternative 1, these areas would not be impacted by any of the various implementation alternatives, the Hybrid Alternative, or the preferred alternative.

The overseas subalternatives would have no impact on cultural resources, as these subalternatives involve no use of DOE management sites. Similarly, the No Action Alternative would have no impact for the same reason.

4.8.7 Air Quality

While all possible precautions and safeguards would be utilized in an effort to conserve air quality, it would be impacted by U.S. acceptance of foreign research reactor spent nuclear fuel.

Basic Implementation of Management Alternative 1: The basic implementation of Management Alternative 1 would not be expected to have major impacts on air quality, and projected emissions from foreign research reactor spent nuclear fuel storage at management sites would not violate Federal or State standards. Dust from construction activities could be controlled with standard techniques. Particulate emissions could have temporary effects on localized visibility, but would not adversely affect Federal or State attainment standards.

Implementation Alternatives: Air quality would be most affected under Implementation Alternative 6, the Hybrid Alternative, or the preferred alternative, all of which involve the use of some form of processing in the United States. Chemical separation would yield a higher effect on air quality than any of the other implementation alternatives.

Since the two overseas subalternatives deal strictly with spent nuclear fuel management overseas, and the No Action Alternative involves no action on the part of DOE or the Department of State, air quality in the United States would not be affected under these alternatives.

4.9 Costs

The costs of implementing various scenarios of the proposed action, including the preferred alternative, plus disposal are presented in this section. Additional details pertaining to costs are provided in Appendix F, Section F.7. For the purpose of the cost analysis, the alternatives described in Section 2.1 were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995c) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. DOE selected six scenarios, including the preferred alternative, for cost analysis. The costs of disposal were estimated for each scenario and are included in the analysis. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 that would affect the cost to the United States.

All costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. For the preferred alternative, however, a wide range of costs is presented because of the uncertainty associated with the new technology development program. An example of an item covered by "other cost factors" would be the cost growth caused by adverse weather that extends the time required to make shipments of the foreign research reactor spent nuclear fuel. The costs are shown as net present values in a consistent accounting framework.

4.9.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, and are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995c). The six cost scenarios are:

1. *Management Alternative 1 (Storage)* — Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site in new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory in existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* — Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Target Material* — Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

4. *Management Alternative 2* — Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

5. *Management Alternative 3* — Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

6. *Preferred Alternative* - Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 3 or 2 and 3.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

4.9.2 Minimum Program Costs

Table 4-64 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. Costs to manage target material (Scenario 3) could be added to the costs of Scenarios 1, 2, 4, and 5 to produce a minimum program cost. Costs to manage target material are included in the preferred alternative (Scenario 6).

Table 4-64 Minimum Program Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Target Material	35
4. Management Alternative 2	1,250
5. Management Alternative 3	675
6. Preferred Alternative[b]	625-950

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

The schedule for activities in Europe under Scenario 5 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13 year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States.

Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States. These net present values imply that all funds required to pay for the program over its 40-year life are received and placed in a trust fund accruing interest at a 4.9 percent real rate of return. This rate of return is required by the Office of Management and Budget for the year ending February, 1996.

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 6 (preferred alternative) is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, the shipping costs in Scenario 6 include the assumption that only 38 cask shipments would be accepted on the West Coast.

4.9.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table 4-64. Table 4-65 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence.

Table 4-65 Other Cost Factors (Net Present Value, Millions of 1996 Dollars in 1996)

	Cost Factors				
Scenario	*Systems Integration & Logistics*	*Component Risks*	*Non-program Risks*	*3% Discount Rate*	*Range*
1. Management Alternative 1 (Storage)	100	75	35	175	385
2. Management Alternative 1 (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Target Material	5	5	0	25	35
4. Management Alternative 2	100	±500	1000	250	350-1850
5. Management Alternative 3	100	±10	150	75	315-335
6. Preferred Alternative[b,c]	100	75	35	225	435

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *Includes target material*

[c] *It is assumed that risk factors are the same as Management Alternative 1 (Storage)*

The other cost factors summarized in Table 4-65 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, 13 years of possible receipts, dozens of foreign ports, up to ten domestic ports, two U.S. management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in several areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the comprehensiveness of the acceptance criteria for aluminum-clad spent nuclear fuel characterization for dry storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, (e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory). For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the current discount rate required by the Office of Management and Budget for the year ending February, 1996 (4.9 percent real) will be reduced to a more historically representative level (e.g., 3 percent) in some future annual update. The base case costs for management outside the United States are discounted at a 3 percent rate. The use of a high discount rate is particularly risky because 1) revenues are likely to be fixed (in $/kgTM) early in the program while expenses are variable and uncertain, and 2) revenues received from the reactor operators during the 1996 through 2008 shipping period will almost certainly exceed the costs of management activities during that period. Mathematically, the excess revenues are placed in a trust fund that compounds interest at the discount rate. If the discount rate exceeds the rate at which funds are actually likely to compound, then outyear program costs (e.g., disposal) could not be met from the principal and accrued interest in the trust fund. A reduction in the discount rate from 4.9 percent to 3.0 percent has a larger impact on the program than any of the technical or systems integration risks.

4.9.4 Potential Total Costs

Table 4-66 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table 4-66 Potential Total Costs (Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Minimum Program Cost*	*Other Cost Factors (Technical)*	*Other Cost Factors (Discount Rate)*	*Potential Total Cost, No Escalation*	*1% Real Escalation, Cumulative*
1. Management Alternative 1 (Storage)	725/775[a],	210	175	~1,100	+11%
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625	85-145	125	~900	+9%
4. Management Alternative 2	1,250	600-1,600	250	2,100-3,100	+13%
5. Management Alternative 3[c]	675	225-275	75	~1000	+9%
6. Preferred Alternative[b]	625-950	210	225	~1,050-1,400	+10%-11%

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

[c] *The total cost risk to the United States is less than 1/2 the total cost risk because a large portion of the*

Table 4-66 shows that the net present value of the potential total costs of implementing the program completely in the United States, including an estimate of program risks but excluding escalation, range from about $900 million for Scenario 2, to $1.4 billion for Scenario 6. Scenario 5 has similar total program costs as Scenario 2 but higher risks for geologic disposal.

In Scenario 4, costs for storing foreign research reactor spent nuclear fuel overseas are highly speculative. In addition, the overseas storage costs are always higher than the more centralized management alternatives because of the extremely high cost of safely and securely managing and disposing of small quantities of spent nuclear fuel in dozens of countries.

The program costs presented in Tables 4-64, 4-65, and 4-66 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and explicitly or implicitly placed in a trust fund. If payments into the trust fund are deferred, then they must be larger than if they had been received on January 1, 1996. For example, if payments are made in 13 equal annual installments every December 31 over the 1996 through 2008 shipping and receiving period, then the constant-dollar payments must increase by 37 percent. A composite of payment schedules, e.g., 13 years for high-income-economy country reactor operators and pay-as-you-go (for the United States) for all other costs, including other-than-high-income-economy country costs, has the effect of increasing the required constant-dollar payments by as much as 25 to 50 percent.

4.9.5 Cost to the United States

The cost of the proposed policy to the United States depends directly on the type of financing arrangement that DOE would adopt in implementing the policy and the discount rate at which revenues from reactor operators accrue interest. Alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. Briefly, the financing arrangements considered are:

1. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *competitive* fee to high-income-economy countries. This is the financial arrangement in the preferred alternative.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for countries with other-than-high-income economies and charges a *full-cost-recovery* fee to high-income-economy countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive* versus the *full-cost-recovery* fee. The U.S. cost under the second arrangement (*no* fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table 4-67 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except Scenario 3) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 4 or 5 would increase its costs by 3 to 4 percent. Fees of $2,000/kgTM, $5,000/kgTM, $7,500/kgTM, and $10,000/kgTM, including a pass-through of shipping charges (all levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation.

The cost to the United States is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the other-than-high-income-economy countries, including shipping, and 2) the difference between the revenues received for management of high-income-economy country foreign research reactor spent nuclear fuel and the total program cost of managing high-income-economy country foreign research reactor spent nuclear fuel, including shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table 4-67 Costs to the United States for Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (developed countries only) $2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	No Fee[d] Developed Countries	Total (excluding shipping)
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
4. Management Alternative 2[f]	500+							1,250 +	1,750+
5. Management Alternative 3[g]	85	1,500	6,000	225	75	(50)	(175)	300	375
6. Preferred Alternative[e]	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from other-than-high-income-economy countries: 15,000 kgTM; from high-income-economy countries: 100,000kgTM; to Dounreay in Scenario 5: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM, essentially all from high-income-economy countries.*

[b] *Full-cost recovery from high-income-economy countries only. The United States bears the costs of the other-than-high-income-economy countries in these cases.*

[c] *Payable in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States.*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 and 2 is $140 Million. The net present value of shipping to the U.S. only in Scenario 5 is $90 Million. The net present value of shipping in Scenario 6 is $160 million. Adding shipping to the net present value for Scenario 2 and Scenario 5 shows that the total program costs for Scenario 5 are slightly lower.*

[e] *Includes target material*

[f] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Scenario 4 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on their income-economy status, commercial nuclear power infrastructure, and other factors.*

[g] *Revenues paid to the United States include pass-through of shipping charges. Costs to the United States for management in Europe include the cost of blending down the HEU to LEU ($20 million).*

Table 4-67 shows that for minimum discounted program costs and fees charged to high-income-economy country reactor operators levelized over 13 years, costs to the United States for the scenarios could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the other-than-high-income-economy countries (including shipping) adds roughly $100 million more to the cost borne by the United States.

If fees in the $2,000 to $10,000 per kgTM range are established and charged over 13 years, the costs to the United States would be as estimated in Table 4-67 plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table 4-66 shows that technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are uncertain but could be in the same range.

4.10 Foreign Research Reactor Spent Nuclear Fuel Risks and Common Risks

This section compares foreign research reactor spent nuclear fuel program risks to those of common activities, such as smoking, flying, receiving a medical X-ray, and so forth.

4.10.1 Risks in the Proposed Action

Preceding sections in Chapter 4 evaluated the risks from radiological and nonradiological activities and accidents in four segments: marine transport, port activities, ground transport, and site activities.

The highest estimated accident MEI risk to the general public from any of the foreign research reactor spent nuclear fuel implementation alternatives is 0.00015 LCF, as shown earlier in Figure 4-23. This would be an individual who lives at the Oak Ridge Reservation boundary under Implementation Alternative 5, Wet Storage Technology for New Construction. This hypothetical individual's chance of incurring a fatal cancer would be increased by less than two in ten thousand.

The highest estimated incident-free population risk to the general public living near any of the DOE management sites from any of the implementation alternatives is less than one-half LCF, as shown earlier in Figure 4-24. This risk occurs under Implementation Alternative 6, Near Term Chemical Separation in the United States, at the Savannah River Site. This risk would be spread among the roughly 600,000 people who live within 80 km (50 miles) of the Savannah River Site, so the average risk among these people would be less than one in a million.

The population risk to the general public due to radiation exposure during ground transport could be as high as 0.22 LCF, as discussed earlier under several of the implementation alternatives to Management Alternative 1.

Nonradiological fatalities are also unlikely. As a practical matter, the only source of nonradiological fatalities to the public is through a traffic accident with a truck or a train. Since truck or train shipments are about 100 or fewer per year, the likelihood of a crash is not high.

4.10.2 Common Radiological Risks

Table 4-68 presents several typical sources of exposure to radiation from everyday life (DOE, 1993e). The average person in the United States receives about 300 mrem each year from natural sources of radiation and about another 50 mrem from manmade sources of radiation. For example, the largest dose listed in Table 4-68 is the 200 mrem/yr from exposure to naturally-occurring radon gas. This is twice the

100 mrem/yr regulatory limit that would apply to marine workers, port workers, and truck drivers under the proposed action. It is also much higher than the dose any member of the general public would be likely to receive.

Table 4-68 Typical Sources of Radiation, Exposures, and Risks

Source	*Dose Rate (mrem/yr)*	*Risk (LCF/yr)*
Radon	200	0.0001
Internal	39	0.000020
Diagnostic X-rays	39	0.000020
Soil, rocks	28	0.000014
Cosmic rays	27	0.000014
Nuclear medicine	14	0.000007
Nuclear fuel cycle	less than 1	less than 5×10^{-7}
Fallout	less than 0.01	less than 5×10^{-9}

There are also large variations in radiation dose to which people are routinely exposed. For example, people who live at high altitudes receive more radiation dose than people who live at sea level. People who live or work in brick, granite, or marble buildings receive more radiation dose than people who live or work in wooden structures. People who live in well-insulated houses receive more radiation dose from trapped radon gas than people who live in well-ventilated houses. Taking all the various factors into account, the annual U.S. dose from background radiation can easily range from 100 mrem for people who live in well-ventilated wooden houses on sandy soil at sea level to about 1000 mrem for people who live in well-insulated houses in the Denver area (de Planque, 1994). Thus, in addition to the average annual radiation dose, routine variations in annual radiation dose are also much larger than the dose any member of the general public would be likely to receive under the proposed action.

4.10.3 Risks from Common Activities

Every activity carries some risk. Table 4-69 shows risks estimated to increase an individual's chance of death in any year by one in one million (Slovic, 1986). For example, a single airline flight across the United States would increase each passenger's radiation dose by about 4 mrem (de Planque, 1994). Most of these voluntary activities would not be considered unusually risky actions, and they can be compared to the risks presented earlier in this chapter for perspective.

Table 4-69 Risks Estimated to Increase Chance of Death in Any Year by One Chance in a Million

Activity	*Cause of Death*
Smoking 1.4 cigarettes	Cancer; heart disease
Living 2 days in New York or Boston	Air pollution
Traveling 16 km (10 mi) by bicycle	Accident
Flying 1,600 km (1,000 mi) by jet	Accident
Living 2 months in Denver on vacation from New York	Cancer caused by cosmic radiation
One chest X-ray taken in a good hospital	Cancer caused by radiation
Drinking 30 12-oz cans of diet soda	Cancer caused by saccharin

5. Applicable Laws, Regulations, and Other Requirements

5.1 Consultation

Certain Federal laws, such as the Endangered Species Act, the Fish and Wildlife Coordination Act, and the National Historic Preservation Act, require consultation and coordination by the United States Department of Energy (DOE) with other governmental entities. These consultation and coordination requirements will commence and be completed as site-specific spent nuclear fuel management projects and decisions are proposed. Any site-specific required consultations will be addressed in the site-specific Environmental Impact Statement (EIS) and/or in Volume I of DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Draft EIS (Table 5-1).

5.2 Laws and Other Requirements

This section identifies and summarizes the major laws, regulations, Executive Orders, and DOE Orders that may apply to the receipt and management of spent nuclear fuel from foreign research reactors.

Section 5.2.1 discusses the major Federal statutes that impose environmental protection and compliance requirements upon DOE. In addition, there may be other State and local measures applicable to the foreign research reactor spent nuclear fuel because Federal law delegates enforcement or implementation authority to State or local agencies. These state- and local-specific requirements are addressed in the site-specific appendices. Section 5.2.2 addresses environmentally-related Executive Orders that clarify issues of national policy and set guidelines under which Federal agencies, including DOE, must act. DOE implements its responsibilities for protection of public health, safety, and the environment through a series of Departmental Orders that are mandatory for operating contractors of DOE-owned facilities. Section 5.2.3 discusses those DOE orders related to environmental, health, and safety protection. Hazardous and radioactive materials transportation regulations are summarized in Section 5.4.2.

5.2.1 Federal Environmental Statutes and Regulations

National Environmental Policy Act (NEPA) of 1969, as amended (42 USC §4321 et seq.)

NEPA establishes a national policy promoting awareness of the environmental consequences of the activity of humans on the environment and also promoting consideration of the environmental impacts during the planning and decision making stages of a project. This Act requires all Federal agencies to prepare a detailed statement on the environmental effects of proposed major Federal actions that may significantly affect the quality of the human environment.

This EIS has been prepared in response to these NEPA requirements and policies. It discusses reasonable alternatives and their potential environmental consequences, and has been prepared in accordance with the Council on Environmental Quality and DOE regulations for implementing the procedural provisions of the NEPA Implementing Procedures (40 CFR Parts 1500 through 1508) and DOE NEPA Implementing Procedures (10 CFR Part 1021).

Table 5-1 Agency Consultations

Subject Area	*Legislation*	*Agency*
Endangered Species	Endangered Species Act of 1973, as amended; State laws	U.S. Fish and Wildlife Service, State agencies
Migratory birds	Migratory Bird Treaty Act	U.S. Fish and Wildlife Service
Bald and Golden eagles	Bald and Golden Eagle Protection Act	U.S. Fish and Wildlife Service
Archaeological, historical, and cultural preservation	National Historic Preservation Act of 1966, Archaeological Resources Protection Act, Antiquities Act, American Indian Religious Freedom Act of 1978, Native American Grave Protection and Repatriation Act of 1990	State Historic Preservation Office, President's Advisory Council, Tribes
Discharge of pollutants to water	Clean Water Act, Safe Drinking Water Act	U.S. Environmental Protection Agency, State agencies
Work in navigable U.S. waters	Clean Water Act, Rivers and Harbors Act, Coastal Management Act	U.S. Army Corps of Engineers
Prime and unique farmlands	Farmland Protection Policy Act of 1981	Soil Conservation Service
Floodplains	Executive Order 11988, Fish and Wildlife Coordination Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, State agencies
Wetlands	Executive Order 11990, Fish and Wildlife Coordination Act, Clean Water Act	U.S. Army Corps of Engineers, U.S. Fish and Wildlife Service, U.S. Environmental Protection Agency, State agencies
Environmental justice	Executive Order 12898	U.S. Environmental Protection Agency
Water body alteration	Fish and Wildlife Coordination Act	U.S. Fish and Wildlife Service, State agencies
River status	Wild and Scenic Rivers Act, Anadromous Fish Conservation Act, Hanford Reach Study Act	U.S. Department of the Interior
Air pollution	Clean Air Act	U.S. Environmental Protection Agency, State and local agencies
Water use and availability	Water Resources Planning Act of 1965, Safe Drinking Water Act, and others	U.S. Environmental Protection Agency, Office of Water Policy, State agencies
Noise	Noise Pollution and Abatement Act of 1970, Noise Control Act of 1972	U.S. Environmental Protection Agency, State agencies
Siting and planning	State siting acts, county zoning regulations	State and County agencies
Waste management and transportation	Solid Waste Disposal Act, as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984; Comprehensive Environmental Response, Compensation, and Liability Act; Emergency Planning and Community Right to Know Act; Hazardous Materials Transportation Act	U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State agencies
Emergency Management & Response	Defense Production Act of 1950, Robert T. Stafford Disaster Relief and Emergency, Assistance Act, National Security Act of 1947	Federal Emergency Management Agency, U.S. Environmental Protection Agency, U.S. Department of Transportation, U.S. Coast Guard, State and local agencies

Atomic Energy Act of 1954, as amended (42 USC §2011 et seq.)

The Atomic Energy Act of 1954 authorizes DOE to establish standards to protect health or minimize dangers to life or property with respect to activities under its jurisdiction. Through a series of DOE Orders, DOE has established an extensive system of standards and requirements to ensure safe operation of its facilities.

Clean Air Act, as amended (42 USC §7401 et seq.)

The Clean Air Act, as amended, is intended to "protect and enhance the quality of the Nation's air resources so as to promote the public health and welfare and the productive capacity of its population." Section 118 of the Clean Air Act, as amended, requires that each Federal agency, such as DOE, with jurisdiction over any property or facility that might result in the discharge of air pollutants, comply with "all Federal, State, interstate, and local requirements" with regard to the control and abatement of air pollution.

The Act requires the Environmental Protection Agency to establish National Ambient Air Quality Standards as necessary to protect public health, with an adequate margin of safety, from any known or anticipated adverse effects of a regulated pollutant (42 USC §7409). The Act also requires establishment of national standards of performance for new or modified stationary sources of atmospheric pollutants (42 USC §7411) and requires specific emission increases to be evaluated so as to prevent a significant deterioration in air quality (42 USC §7470). Hazardous air pollutants, including radionuclides, are regulated separately (42 USC §7412). Air emissions are regulated by the Environmental Protection Agency in 40 CFR Parts 50 through 99. In particular, radionuclide emissions are regulated under the National Emission Standard for Hazardous Air Pollutants Program (see 40 CFR Part 61).

Safe Drinking Water Act, as amended [42 USC §300 (F) et seq.]

The primary objective of the Safe Drinking Water Act, as amended, is to protect the quality of the public water supplies and all sources of drinking water. The implementing regulations, administered by the Environmental Protection Agency unless delegated to the States, establish standards applicable to public water systems. They promulgate maximum contaminant levels (including those for radioactivity), in public water systems, which are defined as water systems that serve at least 15 service connections used by year-round residents or regularly serve at least 25 year-round residents. Safe Drinking Water Act requirements have been promulgated by the Environmental Protection Agency in 40 CFR Parts 100 through 149. For radioactive material, the regulations specify that the average annual concentration of manmade radionuclides in drinking water as delivered to the user by such a system shall not produce a dose equivalent to the total body or an internal organ greater than four mrem per year beta activity. Other programs established by the Safe Drinking Water Act include the Sole Source Aquifer Program, the Wellhead Protection Program, and the Underground Injection Control Program.

Clean Water Act, as amended (33 USC §1251 et seq.)

The Clean Water Act, which amended the Federal Water Pollution Control Act, was enacted to "restore and maintain the chemical, physical and biological integrity of the Nation's water." The Clean Water Act prohibits the "discharge of toxic pollutants in toxic amounts" to navigable waters of the United States. Section 313 of the Clean Water Act, as amended, requires all branches of the Federal Government engaged in any activity that might result in a discharge or runoff of pollutants to surface waters to comply with Federal, State, interstate, and local requirements.

In addition to setting water quality standards for the Nation's waterways, the Clean Water Act supplies guidelines and limitations for effluent discharges from point-source discharges and provides authority for the Environmental Protection Agency to implement the National Pollutant Discharge Elimination System permitting program. The National Pollutant Discharge Elimination System program is administered by the Water Management Division of the Environmental Protection Agency pursuant to regulations in 40 CFR Part 122 et seq. Idaho has not applied for National Pollutant Discharge Elimination System authority from the Environmental Protection Agency. Thus, all National Pollutant Discharge Elimination System permits required for the Idaho National Engineering Laboratory are obtained by DOE through Environmental Protection Agency Region 10 (40 CFR Part 122 et seq.).

Sections 401 and 405 of the Water Quality Act of 1987 added Section 402(p) to the Clean Water Act. Section 402(p) requires that the Environmental Protection Agency establish regulations for issuing permits for stormwater discharges associated with industrial activity. Although any stormwater discharge associated with industrial activity requires a National Pollutant Discharge Elimination System permit application, regulations implementing a separate stormwater permit application process have not yet been adopted by the Environmental Protection Agency.

Resource Conservation and Recovery Act, as amended (Solid Waste Disposal Act) (42 USC §6901 et seq.)

The treatment, storage, or disposal of hazardous and nonhazardous waste is regulated under the Solid Waste Disposal Act as amended by the Resource Conservation and Recovery Act and the Hazardous and Solid Waste Amendments of 1984. Pursuant to Section 3006 of the Act, any State that seeks to administer and enforce a hazardous waste program pursuant to the Resource Conservation and Recovery Act may apply for Environmental Protection Agency authorization of its program. The Environmental Protection Agency regulations implementing the Resource Conservation and Recovery Act are found in 40 CFR Parts 260 through 280. These regulations define hazardous wastes and specify hazardous waste transportation, handling, treatment, storage, and disposal requirements.

The regulations imposed on a generator or a treatment, storage, and/or disposal facility vary according to the type and quantity of material or waste generated, treated, stored, and/or disposed of. The method of treatment, storage, and/or disposal also impacts the extent and complexity of the requirements.

Current Status of Spent Nuclear Fuel under the Resource Conservation and Recovery Act

Historically, DOE chemically reprocessed spent nuclear fuel to recover valuable products and fissionable materials, and as such, the spent nuclear fuel was not a solid waste under the Resource Conservation and Recovery Act.

World events have resulted in significant changes in DOE's direction and operations. In particular, in April 1992, DOE announced the phase-out of reprocessing for the recovery of special nuclear materials. With these changes, DOE's focus on most of its spent nuclear fuel has changed from reprocessing and recovery of materials to storage and ultimate disposition. This in turn has created uncertainty regarding the regulatory status of some of DOE's spent nuclear fuel relative to the Resource Conservation and Recovery Act.

DOE has initiated discussion with the Environmental Protection Agency on the potential applicability of the Resource Conservation and Recovery Act to spent nuclear fuel. Further discussions with Environmental Protection Agency Headquarters and regional offices and State regulators are ongoing to develop a strategy for meeting any the Resource Conservation and Recovery Act requirements that might apply.

Federal Facility Compliance Act (42 USC §6921 et seq.)

The Federal Facility Compliance Act, enacted on October 6, 1992, waives sovereign immunity for fines and penalties for Resource Conservation Recovery Act violations at Federal facilities. However, a provision postpones fines and penalties after 3 years for mixed waste storage prohibition violations at

DOE sites and requires DOE to prepare plans for developing the required treatment capacity for mixed waste stored or generated at each facility. Each plan must be approved by the host State or the Environmental Protection Agency, after consultation with other affected States, and a consent order must be issued by the regulator requiring compliance with the plan. The Federal Facility Compliance Act further provides that DOE will not be subject to fines and penalties for land disposal restriction storage prohibition violations for mixed waste as long as it is in compliance with such an approved plan and consent order and meets all other applicable regulations. This would only apply to foreign research reactor spent nuclear fuel if the Resource Conservation and Recovery Act would apply to storage and treatment of foreign research reactor spent nuclear fuel.

National Historic Preservation Act, as amended (16 USC §470 et seq.)

The National Historic Preservation Act, as amended, provides that sites with significant national historic value be placed on the *National Register of Historic Places*. There are no permits or certifications required under the Act. However, if a particular Federal activity may impact a historic property resource, consultation with the Advisory Council on Historic Preservation will usually generate a Memorandum of Agreement, including stipulations that must be followed to minimize adverse impacts. Coordination with the State Historic Preservation officer is also undertaken to ensure that potentially significant sites are properly identified and appropriate mitigative actions are implemented.

Archaeological Resource Protection Act, as amended (16 USC §470aa et seq.)

This Act requires a permit for any excavation or removal of archaeological resources from public or Native American lands. Excavations must be undertaken for the purpose of furthering archaeological knowledge in the public interest, and resources removed are to remain the property of the United States. Consent must be obtained from the Indian Tribe owning lands on which a resource is located before a permit is issued, and the permit must contain terms or conditions requested by the Tribe.

Native American Grave Protection and Repatriation Act of 1990 (25 USC §3001)

This law directs the Secretary of Interior to assume responsibilities for repatriation of Federal archaeological collections and collections held by museums receiving Federal funding that are culturally affiliated with Native American Tribes. Major actions to be taken under this law include (a) establishing a review committee with monitoring and policy-making responsibilities, (b) developing regulations for repatriation, including procedures for identifying lineal descent or cultural affiliation needed for claims, (c) oversight of museum programs designed to meet the inventory requirements and deadlines of this law, and (d) developing procedures to handle unexpected discoveries of graves or grave goods during activities on Federal or tribal land.

American Indian Religious Freedom Act of 1978 (42 USC §1996)

This Act reaffirms Native American religious freedom under the First Amendment, and sets U.S. policy to protect and preserve the inherent and constitutional right of Native Americans to believe, express, and exercise their traditional religions. The Act requires that Federal actions avoid interfering with access to sacred locations and traditional resources that are integral to the practice of religions.

Religious Freedom Restoration Act of 1993 (42 USC §2000bb et seq.)

This Act prohibits the Government, including Federal Departments, from substantially burdening the exercise of religion unless the Government demonstrates a compelling governmental interest, and the action furthers a compelling Government interest and is the least restrictive means of furthering that interest.

Endangered Species Act, as amended (16 USC §1531 et seq.)

The Endangered Species Act, as amended, is intended to prevent the further decline of endangered and threatened species and to restore these species and their habitats. The Act is jointly administered by the United States Departments of Commerce and the Interior. Section 7 of the Act requires consultation with the U.S. Fish and Wildlife Service to determine whether endangered and threatened species or their critical habitats are known to be in the vicinity of the proposed action. The Idaho National Engineering Laboratory has commenced the consultation process with the U.S. Fish and Wildlife Service (DOE, 1995c). The Savannah River Site, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site have also commenced consultations with the U.S. Fish and Wildlife Service.

Migratory Bird Treaty Act, as amended (16 USC §703 et seq.)

The Migratory Bird Treaty Act, as amended, is intended to protect birds that have common migration patterns between the United States and Canada, Mexico, Japan, and Russia. It regulates the harvest of migratory birds by specifying things such as the mode of harvest, hunting seasons, and bag limits. The Act stipulates that it is unlawful at any time, by any means, or in any manner to "kill . . . any migratory bird." Although no permit for this project is required under the Act, DOE is required to consult with the U.S. Fish and Wildlife Service regarding impacts to migratory birds and to evaluate ways to avoid or minimize these effects in accordance with the U.S. Fish and Wildlife Service Mitigation Policy.

Bald and Golden Eagle Protection Act, as amended (16 USC §668-668d)

The Bald and Golden Eagle Protection Act makes it unlawful to take, pursue, molest, or disturb bald (American) and golden eagles, their nests, or their eggs anywhere in the United States (Sections 668, 668c). A permit must be obtained from the U.S. Department of the Interior to relocate a nest that interferes with resource development or recovery operations.

Wild and Scenic Rivers Act, as amended (16 USC 1271 et seq. 71:8301 et seq.)

The Wild and Scenic Rivers Act, as amended, protects certain selected rivers of the Nation that possess outstanding scenic, recreational, geological, fish and wildlife, historical, cultural, or other similar values. These rivers are to be preserved in a free-flowing condition to protect water quality and other vital national conservation purposes. The purpose of the Act is to institute a national wild and scenic rivers system, to designate the initial rivers that are a part of that system, and to develop standards for the addition of new rivers in the future.

Occupational Safety and Health Act of 1970, as amended (29 USC §651 et seq.)

The Occupational Safety and Health Act establishes standards to enhance safe and healthful working conditions in places of employment throughout the United States. The Act is administered and enforced by the Occupational Safety and Health Administration, a U.S. Department of Labor agency. While the Occupational Safety and Health Administration and Environmental Protection Agency both have a mandate to reduce exposures to toxic substances, the Occupational Safety and Health Administration's jurisdiction is limited to safety and health conditions that exist in the workplace environment. In general, under the Act, it is the duty of each employer to furnish all employees a place of employment free of recognized hazards likely to cause death or serious physical harm. Employees have a duty to comply with the occupational safety and health standards and all rules, regulations, and orders issued under the Act. the Occupational Safety and Health Administration regulations (29 CFR) establish specific standards telling employers what must be done to achieve a safe and healthful working environment. DOE places emphasis on compliance with these regulations at its facilities and prescribes through DOE Orders the Occupational Safety and Health Act standards that contractors shall meet, as applicable to their work at Government-owned, contractor-operated facilities (DOE Order 5480.1B, 5483.1A). DOE keeps and makes available the various records of minor illnesses, injuries, and work-related deaths as required by the Occupational Safety and Health Administration regulations.

Noise Control Act of 1972, as amended (42 USC §4901 et seq.)

Section 4 of the Noise Control Act of 1972, as amended, directs all Federal agencies to carry out "to the fullest extent within their authority" programs within their jurisdictions in a manner that furthers a national policy of promoting an environment free from noise that jeopardizes health and welfare.

5.2.2 Executive Orders

Executive Order 11514 (Protection and Enhancement of Environmental Quality)

Executive Order 11514 requires Federal agencies to continually monitor and control their activities to protect and enhance the quality of the environment and to develop procedures to ensure the fullest practicable provision of timely public information and understanding of the Federal plans and programs with environmental impact to obtain the views of interested parties. The DOE has issued regulations (10 CFR 1021) and DOE Order 5440.1E for compliance with this Executive Order.

Executive Order 11988 (Floodplain Management)

Executive Order 11988 requires Federal agencies to establish procedures to ensure that the potential effects of flood hazards and floodplain management are considered for any action undertaken in a floodplain and that floodplain impacts be avoided to the extent practicable.

Executive Order 11990 (Protection of Wetlands)

Executive Order 11990 requires Governmental agencies to avoid any short- and long-term adverse impacts on wetlands wherever there is a practicable alternative.

Executive Order 12856 (Right-to-Know Laws and Pollution Prevention Requirements)

Executive Order 12856 requires all Federal agencies to reduce the toxic chemicals entering any waste stream. This order also requires Federal agencies to report toxic chemicals entering waste streams; improve emergency planning, response, and accident notification; and encourage clean technologies and testing of innovative prevention technologies.

Executive Order 12898 (Environmental Justice)

Executive Order 12898 requires Federal agencies to identify and address disproportionately high and adverse human health or environmental effects of its programs, policies, and activities on minority and low-income populations.

Table 5-2 DOE Orders Relevant to the DOE Spent Nuclear Fuel Management Program

DOE Order	Subject
1300.2A	Department of Energy Technical Standards Program (5-19-92)
1360.2B	Unclassified Computer Security Program (5-18-92)
1540.2	Hazardous Material Packaging for Transport-Administrative Procedures (9-30-86; Chg. 1. 12-19-88)
3790.1B	Federal Employee Occupational Safety and Health Program (1-7-93)
4330.4A	Maintenance Management Program (10-17-90)
4700.1	Project Management System (3-6-87)
5000.3B	Occurrence Reporting and Utilization of Operations Information (4-9-92)
5400.1	General Environmental Protection Program (11-9-88; Chg. 1, 6-29-90)
5400.2A	Environmental Compliance Issue Coordination (Errata 1-31-89)
5400.4	Comprehensive Environmental Response, Compensation, and Liability Act Requirements (10-6-89)
5400.5	Radiation Protection of the Public and the Environment (2-8-90; Chg. 2, 1-7-93)
5440.1E	National Environmental Policy Act Compliance Program (11-10-92)
5480.1B	Environmental, Safety and Health Program for DOE Operations (9-23-86; Chg. 4, 3-27-90)
5480.3	Environmental Requirements for the Packaging and Transportation of Hazardous Materials, Hazardous Substances, and Hazardous Wastes (7-9-85)
5480.4	Environmental Protection, Safety, and Health Protection Standards (5-15-84; Chg. 4, 1-7-93)
5480.6	Safety of Department of Energy-Owned Nuclear Reactors (9-23-86)
5480.7A	Fire Protection (2-17-93)
5480.8A	Contractor Occupational Medical Program (6-26-92)
5480.9	Construction Safety and Health Program (11-18-87)
5480.10	Contractor Industrial Hygiene Program (6-26-85)
5480.11	Radiation Protection for Occupational Workers (12-21-88; Chg. 2, 6-29-90)
5480.15	Department of Energy Laboratory Accreditation Program for Personnel Dosimetry (12-14-87)
5480.17	Site Safety Representatives (10-05-88))
5480.18A	Accreditation of Performance-Based Training for Category A Reactors and Nuclear Facilities (07-19-91)
5480.19	Conduct of Operations Requirements for DOE Facilities (7-9-90; Chg. 1, 5-18-92)
5480.20	Personnel Selection, Qualification, Training, and Staffing Requirements at DOE Reactor and Nonreactor Nuclear Facilities (2-20-91)
5480.21	Unreviewed Safety Questions (12-24-91)
5480.22	Technical Safety Requirements (2-25-92; Chg. 1, 9-15-92)
5480.23	Nuclear Safety Analysis Reports (4-10-92)
5480.24	Nuclear Criticality Safety (8-12-92)
5480.27	Equipment Qualification for Reactor and Nonreactor Nuclear Facilities (1-15-93)
5480.28	Natural Phenomena Hazards Mitigation (1-15-93)
5480.31	Startup and Restart of Nuclear Facilities (9-15-93)
5481.1B	Safety Analysis and Review System (9-23-86; Chg. 1, 5-19-87)
5482.1B	Environment, Safety, and Health Appraisal Program (9-23-86; Chg. 1, 11-18-91)
5483.1A	Occupational Safety and Health Program for DOE Contractor Employees at Government-Owned, Contractor-Operated Facilities (6-22-83)
5484.1	Environmental Protection, Safety, and Health Protection Information Reporting Requirements (2-21-81; Chg. 7, 10-17-90)
5500.1B	Emergency Management System (4-30-91; Chg. 1, 4-30-91)
5500.2B	Emergency Categories, Classes, and Notification and Reporting Requirements (4-30-91; Chg. 1, 2-27-92)
5500.3A	Planning and Preparedness for Operational Emergencies (4-30-91; Chg. 1, 2-27-92)
5500.4A	Public Affairs Policy and Planning Requirements for Emergencies (6-8-92)
5500.7B	Emergency Operating Records Protection Program (10-23-91)
5500.10	Emergency Readiness Assurance Program (4-30-91; Chg. 1, 2-27-92)
5530.3	Radiological Assistance Program (01-14-92; Change 1, 4-10-92)
5530.5	Federal Radiological Monitoring and Assessment Center (7-10-92)
5630.11A	Safeguards and Security Program (12-7-92)
5630.12A	Safeguards and Security Inspection and Evaluation Program (6-23-92)
5700.6C	Quality Assurance (8-21-91)
5820.2A	Radioactive Waste Management (9-26-88)
6430.1A	General Design Criteria (4-6-89)

5.2.3 DOE Regulations and Orders

Through the authority of the Atomic Energy Act, DOE is responsible for establishing a comprehensive health, safety, and environmental program for its facilities. The regulatory mechanisms through which DOE manages its facilities are the promulgation of regulations and the issuance of DOE Orders.

The DOE regulations are generally found in 10 CFR. These regulations address such areas as energy conservation, administrative requirements and procedures, nuclear safety, and classified information. For the purposes of this EIS, relevant regulations include 10 CFR Part 834, Radiation Protection of the Public and the Environment; 10 CFR Part 835, Occupational Radiation Protection; 10 CFR Part 1021, Compliance with NEPA; and 10 CFR Part 1022, Compliance with Floodplains/Wetlands Environmental Review Requirements. DOE has enacted occupational radiation protection standards to protect DOE and its contractor employees. These standards are set forth in 10 CFR Part 83b, Occupational Radiation Protection. The rules in this part establish radiation protection standards, limits, and program requirements for protecting individuals from ionizing radiation resulting from the conduct of DOE activities, including those conducted by DOE contractors. The activity may be, but is not limited to, design, construction, or operation of DOE facilities. These regulations would be in effect for the construction and operation of any facilities associated with the management of foreign research reactor spent nuclear fuel.

DOE Orders generally set forth policy and the programs and internal procedures for implementing those policies. The major DOE Orders pertaining to the eventual construction and operation of spent nuclear fuel facilities within the DOE Complex are listed in Table 5-2.

5.2.4 Nuclear Regulatory Commission (NRC) Licensing Standards

DOE is proceeding with actions to implement safe, efficient, and cost-effective interim storage of its spent nuclear fuel before final disposition. The need for interim storage has led DOE to evaluate storage technologies and alternative management strategies to provide an optimum solution to storage challenges. Several commercial storage technologies under evaluation for DOE-owned spent nuclear fuel have been licensed and regulated by NRC. In addition, DOE-owned spent nuclear fuel could eventually come under the jurisdiction of NRC if it is to be disposed of in a geological repository. Therefore, DOE is considering having any new interim storage facilities reviewed to determine whether they could meet NRC licensing standards. This approach, if implemented, would provide a testing ground for the development of the technical and administrative protocols between NRC and DOE in the event that some type of NRC regulatory oversight occurs in the future.

5.3 International Regulations

Regulations of the International Atomic Energy Agency

The International Atomic Energy Agency is an agency of the United Nations headquartered in Vienna, Austria. The International Atomic Energy Agency establishes standards for radioactive materials transportation. These are published as model regulations (Safety Series No. 6) that may be adopted by individual nations. These model regulations are regularly revised and updated. Safety Series 6 was revised in 1990 (IAEA, 1990). To the extent considered feasible, the U.S. Nuclear Regulatory Commission (NRC) and the Department of Transportation both periodically review and revise their regulations to bring them into general accord with the International Atomic Energy Agency regulations.

The emphasis of the International Atomic Energy Agency model regulations is on package integrity. To that end, packagings must be shown to survive a hypothetical accident sequence that includes impact, crush, puncture, fire, and immersion. The level of protection is defined by the nature of the contents. The intent of the regulations is to maximize the shipper's contribution to safety, and the shipper (consignor) must certify "that the contents of this consignment are properly described by name; are properly packaged, marked and labeled; and are in proper condition for transport ... " (IAEA, 1990a). The carrier is responsible for following rules for stowage and for segregation from persons.

International Maritime Organization Regulations

The International Maritime Organization publishes the International Maritime Dangerous Goods Code (IMO, 1994), which was developed to supplement the provisions of the 1960 International Convention on the Safety of Life at Sea, as amended, (IMO, 1992) to which the United States is a signatory. Included are regulations that deal with carriage of radioactive material (Class 7 materials). They are based on the International Atomic Energy Agency regulations and deal with segregation of radioactive materials packages from other dangerous goods and other aspects of stowage.

5.4 Domestic Regulations for Radioactive Material Packaging and Transportation

Hazardous and Radioactive Materials Transportation Regulations

Transportation of hazardous and radioactive materials, substances, and wastes are governed by the Department of Transportation, NRC, and the Environmental Protection Agency regulations. These regulations may be found in 49 CFR Parts 171 through 178, 49 CFR Parts 383 through 397, 10 CFR Part 71, and 40 CFR Parts 262 and 265, respectively.

Department of Transportation regulations contain requirements for identifying a material as hazardous or radioactive. These regulations interface with NRC or the Environmental Protection Agency regulations for identifying material, but the Department of Transportation hazardous material regulations govern the hazard communication (such as marking, hazard labeling, vehicle placarding, and emergency response telephone number) and shipping requirements (such as required entries on shipping papers or the Environmental Protection Agency waste manifests).

NRC regulations applicable to radioactive materials transportation are found in 10 CFR Part 71, which includes detailed packaging design requirements and package certification testing requirements. Complete documentation of design and safety analysis and results of the required testing are submitted to the NRC to certify the package for use. This certification testing involves the following components: heat, physical drop onto an unyielding surface, water submersion, puncture by dropping package onto a steel bar, and gas tightness. The recent revision of 10 CFR Part 71, issued on September 28, 1995 (60 CFR 50248), is intended primarily to bring this regulation into conformance with current International Atomic Energy Agency regulations. Revised regulations applicable to the transportation of spent nuclear fuel from foreign research reactors are essentially unchanged.

The Environmental Protection Agency regulations pertaining to hazardous waste transportation are found in 40 CFR Parts 262 and 265. These regulations address labeling and record keeping requirements, including the use of the Environmental Protection Agency waste manifest, which is the required shipping paper for transporting the Resource Conservation and Recovery Act hazardous waste.

5.4.1 NRC Packaging Certification

An NRC certificate is issued as evidence that a packaging and its contents meet applicable Federal regulations. The certificate is issued on the basis of a Safety Analysis Report on the packaging design. Type B packaging must survive certain severe hypothetical accident conditions of impact, puncture, fire, and immersion. The tests are not intended to duplicate accident environments, but rather to produce damage equivalent to extreme accidents. The complete accident sequence is described in 10 CFR, Part 71.73.

Test Sequence for Type B Packagings

The effects of the tests on a package may be evaluated either by subjecting a scale model sample package to the test or by other methods acceptable to the NRC. NRC Regulatory Guide 7.9 allows assessment of package performance by analysis, prototype testing, model testing, or comparison to a similar package. To be judged as surviving, the packaging must not exceed allowable releases defined in 10 CFR 71.51. The dose rate outside the packaging must not exceed 1 rem per hour at a distance of 1 m (3.3 ft) from the packaging surface. The first three tests must be performed on the same package in this order: drop test, puncture test, and thermal test (with an immersion test following for fissile material packagings only).

The drop test consists of a 9-m (30-ft) drop onto a flat, essentially unyielding, horizontal surface, striking the package surface in the position for which maximum damage is expected. An essentially unyielding surface is one that absorbs very little of the energy of impact, which means that the energy of impact is absorbed almost entirely by the package. Unyielding surfaces are constructed of a monolithic concrete base, reinforced by Rebar and covered with a plate of battleship armor. The puncture test consists of a 1-m (40-in) drop onto the upper end of a 15-cm (6-in) solid, vertical, cylindrical bar of mild steel mounted on an essentially unyielding surface. The top of the bar must be horizontal and its edge rounded to a radius of not more than 6 mm (0.25 in).

In the thermal test, the packaging must be exposed for not less than 30 minutes to a heat flux not less than that of a radiation environment of 800°C (1,475°F) with an emissivity coefficient of a least 0.9. The surface absorptivity must be either the value that the package may be expected to possess if exposed to a fire or 0.8, whichever is greater. When it might be significant, convective heat input must be included on the basis of still, ambient air. The packaging may not be artificially cooled after external heat input ceases, and any combustion of materials of construction must be allowed to proceed until it terminates naturally.

Fissile materials packagings for which water in leakage has not been assumed for criticality analysis must be subjected to submersion under a head of water of at least 0.9 m (3 ft) for not less than 8 hours and in the attitude for which the maximum leakage is expected. All packages must be subjected to a separate test in which an undamaged cask is submerged under a head of water of at least 15 m (50 ft) for not less than 8 hours.

Although spent fuel casks have been involved in several accidents, their integrity has never been compromised. The regulatory tests are structured to place an upper bound on the kinds of damage seen in actual severe transportation accidents. Furthermore, after completion of this series of performance qualification tests, Type B packagings are further subjected to a post-accident leak-rate performance test (10 CFR 71.51). In this test, no escape of radioactive material is allowed that exceeds an A2 amount in a week. The A2 amount of an isotope is the maximum activity of that isotope in a potentially dispersible form that is allowed to be shipped in a Type A packaging, which is nonaccident resistant. Safety Series No. 6 lists A2 values for all commonly transported isotopes.

The NRC revised 10 CFR Part 7 regulations governing the transportation of radioactive materials on September 28, 1995 (60 FR 50248). These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than 106 curies be designed and constructed so that its undamaged containment system would withstand an external water pressure of 290 psi, or immersion in 200 meters (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

The use of an essentially unyielding target makes the regulatory certification tests extremely demanding. Real targets are much more yielding. For example, a lead-shield steel cask was dropped 610 m (2,000 ft) from a helicopter onto undisturbed soil (NRC, 1977). Impact velocity was 396 km per hour (235 mph). The cask penetrated 2.4 m (8 ft) into the hard soil but suffered no measurable deformation. An identical cask dropped 9 m (30 ft) onto an essentially unyielding surface during regulatory testing suffered considerably more deformation (Jefferson and Yoshimura, 1978). More recent research has expanded the study of yielding targets (e.g., concrete surfaces) and their comparison with the regulatory surface (Gonzalez et al., 1986).

5.4.2 Transportation Regulations

To assure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR §71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR §71.91. Reports of defects or accidental mishandling must be submitted to NRC.

Communications

Proper communication assists in assuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, and shipping papers or other documents. Labels (49 CFR §172.403) applied to the cask document the contents and the amount of radiation emanating from the cask exterior (transport index). The transport index lists the ionizing radiation level (in mrem/hr) at a distance of 1 m (3.3 ft) from the cask surface.

In addition to the label requirements, markings (49 CFR Subpart D and §173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (40 CFR §172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricator's) name and address, the Certificate of Compliance number, and the gross weight (10 CFR §71.83).

Placards (49 CFR §172.500) are applied to the transport vehicle or freight container holding the transportation cask. The placards indicate the radioactive nature of the contents. In the United States, spent nuclear fuel is a Highway Route Controlled Quantity which must be placarded according to 49 CFR §172.507. Placards provide the first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers should have entries identifying the following: the name of the shipper, emergency response telephone number, description of spent nuclear fuel, and the shipper's certificate as described in 49 CFR §172 Subpart C.

In addition, drivers of motor vehicles transporting spent nuclear fuel must have training in accordance with the requirements of 49 CFR §172.700. The training requirements include: familiarization with the regulations, emergency response information, and the spent nuclear fuel communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

Except for exclusive-use shipments, requirements relating to transport indexes state that:

> ". . . the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and:
>
> (1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and,
>
> (2) Each freight container or group of freight containers is (are) handled and stowed in such a manner that groups are separated from each other by a distance of at least six m (20 ft)," [49 CFR §176.704(c)].

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried onboard a vessel in accordance with the following:

> "(1) The material must be in proper condition for transportation according to the requirements of this subchapter;
>
> (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. However, vertical restraint is not required if the shape of the packages and the stuffing pattern precludes shifting of the load;
>
> (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends;
>
> (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by 49 CFR §172 Subpart F of the Hazardous Materials Regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident, or alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials onboard a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Marine Transport

Relevant regulations applying to transport of spent nuclear fuel by vessel are found in 10 CFR Parts 71 and 73, and 49 CFR Part 176. The USCG, part of the Department of Transportation, inspects vessels for compliance with applicable regulations and requires 24-hour prenotification (33 CFR 160.207, 211, and 213).

Section 49 CFR 171.12 (d) states that: "Radioactive materials being imported into or exported from the United States, or passing through the United States in the course of being shipped between places outside the U.S., may be offered and accepted for transportation when packaged, marked, labeled, and otherwise prepared for shipment in accordance with the IAEA 'Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, 1985 Edition' including 'Supplement 1988.'" Certain specified conditions of this section must be complied with. For example, highway-route-controlled quantities of radioactive material must be shipped in accordance with appropriate provisions of the hazardous materials regulations and a Certificate of Competent Authority must be obtained, with any necessary revalidations. A Certificate of Competent Authority fulfills the International Atomic Energy Agency requirement for multilateral approval for a shipment of Type B packages in international commerce (IAEA, 1990a).

Section 49 CFR 176.5 details the application of the regulations to vessels: "...this subchapter applies to each domestic or foreign vessel when in the navigable waters of the United States, regardless of its character, tonnage, size or service, and whether self-propelled or not, whether arriving or departing, underway, moored, anchored, aground, or while in drydock." Exempted from the regulations are vessels not engaged in commercial service, a vessel used exclusively for pleasure, a vessel of 500 gross tons or smaller, engaged in fisheries, etc. Section 49 CFR 176.15 provides for enforcement of 40 CFR Subchapter C:

> "(a) An enforcement officer of the U.S. Coast Guard may at any time and at any place, within the jurisdiction of the United States, board any vessel for the purpose of enforcement of this subchapter and inspect any shipment of hazardous materials as defined in this subchapter."

Provision is also made in this section to detain a vessel that is in violation of the hazardous materials regulations.

The USCG may accept a certificate of loading issued by the National Cargo Bureau, Inc., as evidence that the cargo is stowed in conformity with law and regulatory requirements. The National Cargo Bureau, Inc., is a non-profit organization directed by government and industry representatives (49 CFR 176.18 authorizes inspectors of the National Cargo Bureau, Inc., to assist the USCG in administering the hazardous materials regulations). Their functions are as follows:

> "(1) Inspection of vessels for suitability for loading hazardous materials; (2) Examination of stowage of hazardous materials; (3) Making recommendations for stowage requirements of hazardous materials cargo; and, (4) Issuance of certificates of loading setting forth that the stowage of hazardous materials is in accordance with the requirements of 46 U.S.C. 170 and its subchapter."

Detailed requirements for shipping radioactive material are located in Part 176 Subpart M of the hazardous materials regulations. General radioactive materials stowage requirements of 49 CFR 176.700 state that: "(b) A package of radioactive materials which in still air has a surface temperature more than 5°C (9°F) above the ambient air may not be overstowed with any other cargo. If the package is stowed under the deck, the hold or compartment in which it is stowed must be ventilated."

Except for exclusive-use shipments, requirements of 176.704 (c) relating to transport indexes state that:

> "the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and: (1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and, (2) Each freight container or group of freight containers is handled and stowed in such a manner that groups are separated from each other by a distance of at least six meters (20 feet)."

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried on board a vessel in accordance with the following:

> "(1) The material must be in proper condition for transportation according to the requirements of this subchapter; (2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. Vertical restraint is not required if the shape of the packages, loading pattern, and horizontal restraint preclude vertical movement of the load within the freight container or transport vehicle; (3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends; (4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by Subpart F of Part 172 of the hazardous materials transportation regulations [49 CFR 176.76(f)].

Section 49 CFR 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident or, alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR 176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials on board a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

Ground Transport

Overland shipments (by rail car or by truck) are regulated by a variety of the Department of Transportation and NRC regulations dealing with packaging, notification, escorts and communication. In addition, there are specific regulations for carriage by truck and carriage by rail.

When provisions are made to secure a package so that its position within the transport vehicle remains fixed during transport, with no loading or unloading between the beginning and end of transport, a package shipped overland in exclusive-use closed transport vehicles may not exceed the following radiation levels as provided in 49 CFR 173.441(b):

1. 200 millirem per hour on the external surface of the package unless the following conditions are met, in which case the limit is 1,000 millirem per hour;

    i. The shipment is made in a closed transport vehicle;

    ii. The package is secured within the vehicle so that its position remains fixed during transportation; and

    iii. There are no loading or unloading operations between the beginning and end of the transportation;

2. 200 millirem per hour at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load (or enclosure is used), and on the lower external surface of the vehicle;

3. 10 millirem per hour at any point 2 m (6.6 ft) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 m (6.6 ft) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle); and

4. 2 millirem per hour in any normally occupied space.

The shipper of record must comply with the requirements of 10 CFR 71.5 and 73.37. Section 71.5 provides that all overland shipments must be in compliance with Department of Transportation and NRC regulations, which provide for security of irradiated reactor fuel. General requirements include:

- Provide notification to NRC in advance of each shipment,
- Develop a shipping plan,
- Provide escort instructions
- Establish a communication center to be staffed 24 hours a day,
- Make arrangements with local law enforcement agencies along the route for their response, if not using law enforcement personnel as escort, ensure that the escorts are trained in accordance with 10 CFR 73.37 Appendix D, and
- Ensure that escorts make notification calls every 2 hours to the communications center.

Additional requirements include having two armed escorts within heavily populated areas (when not in heavily populated areas, only one escort is needed) and the capability of communicating with the communications center and local law enforcement agencies through a radiotelephone or other NRC-approved means of two-way voice communications.

The shipper of record, as required by 49 CFR 173.22, provides physical security measures for spent fuel shipments equivalent to those of the NRC. The shipper and his agent will provide notification for unclassified spent fuel shipments to State officials.

For carriage by truck, the carrier will use interstate highways or State-designated preferred routes for movement of radioactive materials in conformity with the Department of Transportation rule known as Docket HM-164. These regulations, found in 49 CFR 397.101, establish routing and driver training requirements for highway carriers of packages containing "highway-route-controlled quantities" of radioactive materials. Spent fuel shipments constitute such quantities. Department of Transportation rules make those routes designated by appropriate State agencies enforceable by the Federal Government according to the Department of Transportation's own determination that such route designations, when accompanied by an adequate safety analysis, are likely to result in further reduction of radiological risk.

For carriage by rail car, each shipment by the railroad must comply with 49 CFR 174, in particular, 174 Subpart K, Detailed Requirements for Radioactive Materials.

5.5 Emergency Management and Response

5.5.1 Authorities and Directives

Emergency Planning and Community Right-to-Know Act of 1986 (42 USC §11001 et seq.) (also known as "SARA Title III")

Under Subtitle A of this Act, Federal facilities, including those owned by DOE, provide various information (such as inventories of specific chemicals used or stored and releases that occur from these sites) to the State Emergency Response Commission and to the Local Emergency Planning Committee to ensure that emergency plans are sufficient to respond to unplanned releases of hazardous substances. Implementation of the provisions of this Act began voluntarily in 1987, and inventory and annual emissions reporting began in 1988 based on 1987 activities and information. DOE also requires compliance with Title III as a matter of Agency policy. The requirements for this Act were promulgated by the Environmental Protection Agency in 40 CFR Parts 350 through 372.

The Toxic Substances Control Act also regulates the treatment, storage, and disposal of certain toxic substances not regulated by the Resource Conservation and Recovery Act or other statutes, particularly polychlorinated biphenyls, chlorofluorocarbons, and asbestos.

Quantities of Radioactive Materials Requiring Consideration of the Need for an Emergency Plan for Responding to a Release (10 CFR Part 30.72 Schedule C)

This list is the basis for both the public and private sector to determine if the radiological materials they deal with must have an emergency response plan for unscheduled releases. It is one of the threshold criteria documents for DOE Hazards Assessments required by DOE Order 5500.3A, "Planning and Preparedness for Operational Emergencies" (DOE, 1991c).

Occupational Safety and Health Administration Emergency Response, Hazardous Waste Operations and Worker Right to Know (29 CFR)

This regulation sets down the Occupational Safety and Health Administration requirements for employee safety in a variety of working environments. It addresses employee emergency and fire prevention plans (Section 1910.38), hazardous waste operations and emergency response (Section 1910.120), and hazards communication (Section 1910.1200) that enables employees to be aware of the dangers they face from hazardous materials at their workplace.

Emergency Management and Assistance (44 CFR 1.1)

This regulation contains the policies and procedures for the Federal Emergency Management Act, National Flood Insurance Program, Federal Crime Insurance Program, Fire Prevention and Control Program, Disaster Assistance Program, and Preparedness Program including radiological planning and preparedness.

Hazardous Materials Tables & Communications, Emergency Response Information Requirements (49 CFR Part 172)

The regulatory requirements for marking, labeling, placarding, and documenting hazardous materials shipments are defined in this regulation. It also specifies the requirements for providing hazardous material information and training.

Public Law 93-288, as Amended by Public Law 100-707, "Robert T. Stafford Disaster Relief and Emergency Assistance Act," November 23, 1988

The Robert T. Stafford Disaster Relief and Emergency Assistance Act, P.L. 93-288, as amended, provides an orderly and continuing means of assistance by the Federal Government to State and local governments in carrying out their responsibilities to alleviate the suffering and damage resulting from disasters. The President, in response to a State Governor's request, may declare an "emergency" or "major disaster," in order to provide Federal assistance under the Act. The President, in Executive Order 12148, delegated all functions, except those in Sections 301, 401, and 409, to the Director, Federal Emergency Management Agency. The Act provides for the appointment of a Federal Coordinating Officer who will operate in the designated area with a State Coordinating Officer for the purpose of coordinating State and local disaster assistance efforts with those of the Federal Government.

Public Law 96-510, "Comprehensive Environmental Response, Compensation, and Liability Act of 1980," Section 104(i), 42 U.S.C. 9604(i)

More popularly known as "Superfund," this Act provides the needed general authority for Federal and State governments to respond directly to hazardous substances incidents. The Act requires reporting of spills, including radioactive, to the National Response Center.

Public Law 98-473, Justice Assistance Act of 1984

These Department of Justice regulations implement the Emergency Federal Law Enforcement Assistance functions vested in the Attorney General. Those functions were established to assist State and/or local units of government in responding to a law enforcement emergency. The Act defines the term "law enforcement emergency" as an uncommon situation which requires law enforcement, which is or threatens to become of serious or epidemic proportions, and with respect to which State and local resources are inadequate to protect the lives and property of citizens, or to enforce the criminal law. Emergencies that are not of an ongoing or chronic nature, such as the Mount Saint Helens volcanic eruption, are eligible for Federal law enforcement assistance. Such assistance is defined as funds, equipment, training, intelligence information, and personnel. Requests for assistance must be submitted in writing to the Attorney General by the chief executive office of a State. The Plan does not cover the provision of law enforcement assistance. Such assistance will be provided in accordance with the regulations referred to in this paragraph [28 CFR Part 65, implementing the Justice Assistance Act of 1984] or pursuant to any other applicable authority of the Department of Justice.

Communications Act of 1934, as Amended

This Act gives the Federal Communications Commission emergency authority to grant Special Temporary Authority on an expedited basis to operate radio frequency devices.

5.5.2 Executive Orders

Executive Order 10480, as Amended, "Further Providing for the Administration of the Defense Mobilization Program," August 1953

Part II of the Order delegates to the Director, Federal Emergency Management Agency, with authority to redelegate, the priorities and allocation functions conferred on the President by Title I of the Defense Production Act of 1950, as amended.

Executive Order 12148, "Federal Emergency Management," July 20, 1979

Executive Order 12148 transferred functions and responsibilities associated with Federal emergency management to the Director, Federal Emergency Management Agency. The Order assigns the Director, Federal Emergency Management Agency, the responsibility to establish Federal policies for and to coordinate all civil defense and civil emergency planning, management, mitigation, and assistance functions of Executive Agencies.

Executive Order 12472, "Assignment of National Security and Emergency Preparedness Telecommunications Functions," April 3, 1984

Executive Order 12472 establishes the National Communication System. The National Communication System consists of the telecommunications assets of the entities represented on the National Communication System Committee of Principals and an administrative structure consisting of the Executive Agent, the National Communication System Committee of Principals, and the Manager. The National Communication System Committee of Principals consists of representatives from those Federal departments, agencies, or entities, designated by the President, which lease or own telecommunications facilities or services of significance to national security or emergency preparedness.

Executive Order 12656, "Assignment of Emergency Preparedness Responsibilities," November, 1988

This order assigns emergency preparedness responsibilities to Federal departments and agencies.

5.5.3 Emergency Planning Documents

"Federal Radiological Emergency Response Plan," November 1985

This document is to be used by Federal agencies in peacetime radiological emergencies. It primarily concerns the off-site Federal response in support of State and local governments with jurisdiction for the emergency. The Federal Radiological Emergency Response Plan provides the Federal Government's concept of operations based on specific authorities for responding to radiological emergencies, outlines Federal policies and planning assumptions that underlie this concept of operations and on which Federal agency response plans were based, and specifies authorities and responsibilities of each Federal agency that may have a significant role in such emergencies.

"National Plan for Telecommunications Support [in Non-Wartime Emergencies," January 1992

This plan provides guidance in planning for and providing telecommunications support for Federal agencies involved in emergencies, major disasters, and other urgent events, excluding war.

Department of Defense Directive 3025.1, "Military Support to Civil Authorities," 1992

This directive outlines Department of Defense policy on assistance to the civilian sector during disasters and other emergencies. Use of the Department of Defense military resources in civil emergency relief operations will be limited to those resources not immediately required for the execution of the primary defense mission. Normally, the Department of Defense military resources will be committed as a supplement to non-Department of Defense resources that are required to cope with the humanitarian and property protection requirement caused by the emergency. In any emergency, commanders are authorized to employ Department of Defense resources to save lives, prevent human suffering, or mitigate great property loss. Upon declaration of a major disaster under the provisions of P.L. 93-288, as amended, the Secretary of the Army is the Department of Defense Executive Agent, and the Director of Military Support is the action agent for civil emergency relief operations. Military personnel will be under command of and directly responsible to their military superiors and will not be used to enforce or execute civil law in violation of 18 U.S.C. 1385, except as otherwise authorized by law. Military resources shall not be procured, stockpiled, or developed solely to provide assistance to civil authorities during emergencies.

Federal Preparedness Circular 8, "Public Affairs in Emergencies"

This Circular establishes the Interagency Committee on Public Affairs in Emergencies to coordinate public information planning and operations for management of emergency information. The Circular was reviewed in draft by the Interagency Committee on Public Affairs in Emergencies and will receive formal department and agency review.

American Red Cross Disaster Services Regulations and Procedures, ARC 3003, January 1984

This document details the delegation of disaster services program responsibilities to officials and units of the American Red Cross. Also defined are the American Red Cross administrative regulations and procedures for disaster planning, preparedness, and response.

Statement of Understanding between the Federal Emergency Management Agency and the American National Red Cross, January 22, 1982

The statement of understanding between the Federal Emergency Management Act and the American National Red Cross describes major responsibilities in disaster preparedness planning and operations in the event of a war-caused national emergency or a peacetime disaster, outlines areas of mutual support and cooperation, and provides a frame of reference for similar cooperative agreements between State and local governments and the operations headquarters and chapters of the American Red Cross.

6. List of Preparers

Name:	Charles R. Head
Affiliation:	Department of Energy
Education:	MS, Control Theory, George Washington University BS, Electrical Engineering, Rice University
Experience/ Technical Specialty:	Twenty-seven years. Nuclear safety oversight, safeguards and security requirements, Tiger Team assessments, spent fuel storage.
EIS Responsibility:	DOE Foreign Research Reactor Spent Nuclear Fuel Project Manager

Name:	Patrick J. Wells
Affiliation:	Department of Energy
Education:	MS, Engineering Management, George Washington University BS, Civil Engineering, Marquette University
Experience/ Technical Specialty:	Eleven years. Occupational safety and health, oversight system acquisition, reliability, and maintainability.
EIS Responsibility:	Assistant to the DOE Project Manager

Name:	Darren W. Piccirillo
Affiliation:	Department of Energy
Education:	BS, Physics, Southern Connecticut State University
Experience/ Technical Specialty:	Eight years. Reactor plant construction and operation.
EIS Responsibility:	Spent fuel characterization, storage impact analysis

Name:	Eleanor R. Busick
Affiliation:	Department of State
Education:	MA, Economics, Yale University BA, Oberlin College
Experience/ Technical Specialty:	Nineteen years. Nuclear nonproliferation policy, economics (fuel markets, finance).
EIS Responsibility:	Nonproliferation and national policies

Name:	Fred McGoldrick
Affiliation:	Department of State
Education:	PhD, Political Science, American University BA, Boston College
Experience/ Technical Specialty:	Twenty-three years. Nuclear nonproliferation policy.
EIS Responsibility:	Nonproliferation and national policies

Name:	Ibrahim H. Zeitoun
Affiliation:	Science Applications International Corporation
Education:	PhD, Environmental Sciences, Michigan State University MS, Fisheries, Michigan State University BS, Chemistry & Zoology, University of Alexandria
Experience/ Technical Specialty:	Twenty-two years. Waste management, environmental assessment, and NEPA compliance.
EIS Responsibility:	Overall EIS Project Manager. NEPA compliance, spent fuel management environmental justice

Name:	Audrey J. Adamson
Affiliation:	Science Applications International Corporation
Education:	BA, University of Michigan MPA, George Washington University
Experience/ Technical Specialty:	Eleven years. International, environmental, and technology policy analysis, public outreach, and technical writing.
EIS Responsibility:	Emergency response and communication planning

Name:	Douglas H. Amick
Affiliation:	Science Applications International Corporation
Education:	AS, Law Enforcement, Walter State Community College
Experience/ Technical Specialty:	Twenty years. Marine ports, port safety inspections and security, plant operations, transportation risks.
EIS Responsibility:	Port safety, operation and characteristics

Name:	Paula W. Austin
Affiliation:	Science Applications International Corporation
Education:	BS, Management and Technology, University of Maryland
Experience/ Technical Specialty:	Nineteen years. Nuclear and environmental policy analysis, public outreach, technical writing.
EIS Responsibility:	EIS Summary, public hearings task leader

Name:	Rakesh Bahadur
Affiliation:	Science Applications International Corporation
Education:	PhD, Groundwater Hydrology, Colorado State University MS, Groundwater Hydrology, Colorado State University MSc, Geology, Punjab University
Experience/ Technical Specialty:	Fifteen years. Hydrology, site characterization, environmental assessment, and risk assessment.
EIS Responsibility:	Affected environment, Geographic Information Systems, Environmental Justice

Name:	Bruce M. Biwer
Affiliation:	Argonne National Laboratory
Education:	PhD, Chemistry, Princeton University MA, Chemistry, Princeton University BA, Chemistry, St. Anselm College
Experience/ Technical Specialty:	Fourteen years. Radiological pathway analysis, dose calculations, radiological transportation risk analysis.
EIS Responsibility:	Radiological transportation risk analysis

Name:	Keith R. Brown
Affiliation:	Science Applications International Corporation
Education:	MS, Mathematics, Idaho Sate University BS, Mathematics, Idaho State University
Experience/ Technical Speciality:	Thirty-four years. Transportation casks, research reactor operations, and spent nuclear fuel management
EIS Responsibility:	Spent fuel management, transportation assessment

Name:	Burrus M. Carnahan
Affiliation:	Science Applications International Corporation
Education:	LLM, International Law, University of Michigan JD, Northwestern University BA, Political Science, Drake University
Experience/ Technical Specialty:	Twenty-one years. Weapons proliferation, international agreements, and arms control.
EIS Responsibility:	Nonproliferation policies
Name:	Harry Chernoff
Affiliation:	Science Applications International Corporation
Education:	MBA, Marymount University BA, Economics, College of William & Mary
Experience/ Technical Specialty:	Seventeen years. Economics, socioeconomics, finance, and engineering economics.
EIS Responsibility:	Costs, and Socioeconomics
Name:	Louis Cofone, Jr.
Affiliation:	Science Application International Corporation
Education:	MS, Microbiology, University of Dayton BS, Biology, Philadelphia College of Pharmacy and Science
Experience/ Technical Specialty:	Twenty years. Computer information analysis, data base development and management, environmental and health risk assessment.
EIS Responsibility:	Development and management of the comment tracking and document control system
Name:	Cecil C. Cross, III
Affiliation:	Science Applications International Corporation
Education:	MEM, Environmental Management, Duke University BA, Biology, Gettysburg College
Experience/ Technical Specialty:	Thirteen years. Regulatory compliance, environmental assessment, and hazardous and mixed waste management.
EIS Responsibility:	Port operation, port environmental and climatic conditions

Name:	Larry Danese
Affiliation:	Science Applications International Corporation
Education:	MBA, Florida International University BS, Electrical Engineering, University of Florida
Experience/ Technical Specialty:	Twenty-three years. Cask design, transportation systems, emergency response, and regulatory compliance.
EIS Responsibility:	Transportation casks and regulations, port operation regulations and activities

Name:	Gary M. DeMoss
Affiliation:	Science Applications International Corporation
Education:	MS, Engineering Administration, Virginia Polytechnic Institute BS, Mechanical Engineering, University of Virginia
Experience/ Technical Specialty:	Thirteen years. Risk analysis, reliability and safety engineering, uranium enrichment, and transportation.
EIS Responsibility:	Ground transportation safety and impact analysis, port selection and operation

Name:	Scott E. Drummond, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Marine Transportation, SUNY Maritime College
Experience/ Technical Specialty:	Forty-two years. Strategic sealift, logistics support, ocean survey, nautical charting, SWATH ship design and operation.
EIS Responsibility:	Port information

Name:	Habib A. Durrani
Affiliation:	Science Applications International Corporation
Education:	BSc, Engineering Science, Peshawar University
Experience/ Technical Specialty:	Twenty years. Nuclear facilities operation, design maintenance regulations and safety
EIS Responsibility:	Chemical separation technologies

Name:	Barbara M. Ebert
Affiliation:	Science Applications International Corporation
Education:	MA, National Security Studies, Georgetown University BS, Foreign Service, Comparative and Regional Studies, Georgetown University
Experience/ Technical Specialty:	Twelve years. Weapons proliferation.
EIS Responsibility:	Nonproliferation policies

Name:	Martin W. Ebert
Affiliation:	Science Applications International Corporation
Education:	BSc, Nuclear Engineering, University of Arizona MSc, Applied Physics, University of Strathclyde
Experience/ Technical Specialty:	Twenty-four years. Nuclear powerplant operations, spent fuel technology, and technical safety requirements.
EIS Responsibility:	Marine and port safety and impact analysis, public hearings response coordinator

Name:	Daniel W. Gallagher
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, Rensselaer Polytechnic Institute BS, Nuclear Engineering, Rensselaer Polytechnic Institute
Experience/ Technical Specialty:	Fifteen years. Reliability and risk engineering, probabilistic safety assessment, plant design, and regulatory analysis.
EIS Responsibility:	Marine and port safety and impact analysis

Name:	Reginald L. Gotchy
Affiliation:	Science Applications International Corporation
Education:	PhD, Radiation Biology, Colorado State University MS, Radiation Health, Colorado State University BS, Zoology, University of Washington
Experience/ Technical Specialty:	Twenty-six years. NEPA compliance, safety analysis, risk assessment, radiation biology, health physics (Certified Health Physicist), and emergency response planning.
EIS Responsibility:	Port selection and radiological consequences and health effects

Name:	Peter Grier
Affiliation:	Science Applications International Corporation
Education:	BS, Psychology, University of Maryland
Experience/ Technical Specialty:	Twenty years. Emergency management, commercial nuclear energy, quality assurance, and transportation regulatory compliance.
EIS Responsibility:	Emergency response, security, and communication planning

Name:	Timothy T. Holmes
Affiliation:	Science Applications International Corporation
Education:	JD, University of Kansas School of Law BA, Business, Washburn University
Experience/ Technical Specialty:	Four years. Legal and environmental analysis, involving NEPA, RCRA, and other environmental regulations, document review and contract compliance.
EIS Responsibility:	Comment Response Document task leader, technical editor

Name:	Joseph W. James
Affiliation:	Science Applications International Corporation
Education:	BSG, Administration of Justice, American University MA, Management, Central Michigan University PhD, Environmental Science, LaSalle University
Experience/ Technical Specialty:	Thirty years. Nuclear safeguards and security, standards development, quality assurance, regulatory analysis, and licensing support.
EIS Responsibility:	Safeguards and security planning

Name:	Roy Karimi
Affiliation:	Science Applications International Corporation
Education:	ScD, Nuclear Engineering, Massachusetts Institute of Technology NE, Nuclear Engineering, Massachusetts Institute of Technology MS, Nuclear Engineering, Massachusetts Institute of Technology BSc, Chemical Engineering, Abadan Institute of Technology
Experience/ Technical Specialty:	Fourteen years. Nuclear powerplant safety, risk and reliability analysis, design analysis, and probabilistic risk assessment.
EIS Responsibility:	Spent fuel characterization, accident and impact analysis, quality control reviews

Name:	Stephen J. Krill, Jr.
Affiliation:	Science Applications International Corporation
Education:	BS, Nuclear and Power Engineering, University of Cincinnati
Experience/ Technical Specialty:	Five years. Safety and risk analysis, reactor and fuel processing system design, operation and inspection, and emergency preparedness.
EIS Responsibility:	Transportation cask descriptions, environmental consequences

Name:	Merritt E. Langston, PE
Affiliation:	Science Applications International Corporation
Education:	BS, MS, Metallurgical Engineering, Missouri School of Mines
Experience/ Technical Specialty:	Thirty-two years. Quality management, nuclear engineering, defense programs, nuclear waste management. Six years. Reactor containment materials development.
EIS Responsibility:	Technical reviews, quality control task leader

Name:	Christi D. Leigh
Affiliation:	Sandia National Laboratories
Education:	PhD, Engineering, University of New Mexico MS, Chemical Engineering, Stanford University BS, Chemical Engineering, Arizona State University
Experience/ Technical Specialty:	Six years. Radioactive and hazardous waste management and minimization. Five years. Nuclear reactor safety.
EIS Responsibility:	At-sea submerged cask risk assessment

Name:	Charles D. Massey
Affiliation:	Sandia National Laboratories
Education:	PhD, Radiation Health, University of Pittsburgh MS, Health Physics, University of Pittsburgh BS, Marine Transportation, U.S. Merchant Marine Academy
Experience/ Technical Specialty:	Thirteen years. NEPA, risk assessment, transportation and energy technology evaluation.
EIS Responsibility:	Marine transportation risk assessment and impacts

Name:	Ronya J. McMillen
Affiliation:	Science Applications International Corporation
Education:	MA, International Science & Technology Policy, George Washington University BS, Sociology, Chatham College
Experience/ Technical Specialty:	Thirteen years. Technical analysis and report writing. Four years public outreach and policy analysis on domestic and foreign nuclear technology regulatory issues.
EIS Responsibility:	EIS Summary Coordinator, Public comment and hearing summaries

Name:	Charles D. Miller
Affiliation:	Science Applications Internal Corporation
Education:	BS, Marine Transportation, SUNY Maritime College
Experience/ Technical Specialty:	Forty years. Transportation/distribution system design, planning, implementation,.and management.
EIS Responsibilty:	Port information
Name:	Todd Miller
Affiliation:	Science Applications International Corporation
Education:	BS, Civil Engineering, Worcester Polytechnic Institute
Experience/ Technical Specialty:	Four years. Safety analysis, environmental assessment, NEPA compliance.
EIS Responsibility:	Accident analysis, radiological consequences
Name:	Steven M. Mirsky
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, Pennsylvania State University BS, Mechanical Engineering, The Cooper Union
Experience/ Technical Specialty:	Nineteen years. Safety analysis, nuclear powerplant design, operations, and foreign nuclear powerplant system analysis.
EIS Responsibility:	Storage technology, safety and impact analysis
Name:	Frederick A. Monette
Affiliation:	Argonne National Laboratory
Education:	MS, Health Physics, Colorado State University BA, Physics, St. Johns University
Experience/ Technical Specialty:	Six years. Radiological risk assessment, radiological transportation risk analysis, dose calculations.
EIS Responsibility:	Radiological transportation risk and impacts analysis
Name:	Michael Moore
Affiliation:	Science Applications International Corporation
Education:	BA, Economics, University of Maryland
Experience/ Technical Specialty:	Twelve years. Analysis and design of environmental/waste information systems, drafting and editing of technical documents for energy, environmental, and defense initiatives.
EIS Responsibility:	Quality control reviews, technical editor

Name:	Alexander P. Murray
Affiliation:	Science Applications International Corporation
Education:	MS, Chemical Engineering, Carnegie-Mellon University BS, Chemical Engineering, Carnegie-Mellon University
Experience/ Technical Specialty:	Nineteen years. Waste management, environmental and regulatory compliance, design engineering and computer modeling, nuclear reactors and systems, dose analysis, nuclear fuel cycle, and spent fuel reprocessing.
EIS Responsibility:	Storage and chemical separation technologies

Name:	Iral C. Nelson
Affiliation:	Pacific Northwest Laboratories
Education:	MA, Physics, University of Oregon Diplomate, American Board of Health Physics BS, Mathematics, University of Oregon
Experience/ Technical Specialty:	Thirty-nine years. Health physics, radiation protection, and NEPA compliance and reviews.
EIS Responsibility:	Affected environment, environmental consequences

Name:	Aris Papadopoulos
Affiliation:	Science Applications International Corporation
Education:	MS, Nuclear Engineering, University of Utah BS, Physics, Hamline University
Experience/ Technical Specialty:	Twenty-two years. Safety analysis assessment, regulatory reviews, reactor safety, fuel cycle facility systems, radioactive waste management, and accident analysis support.
EIS Responsibility:	Transportation casks, storage alternatives, and impact assessment

Name:	Kathleen Rhoads
Affiliation:	Pacific Northwest Laboratories
Education:	MS, Radiological Sciences, University of Washington BS, Microbiology, University of Washington
Experience/ Technical Specialty:	Nineteen years. Risk assessment, radiation doses, health effects from energy production.
EIS Responsibility:	Radiological consequences analysis

Name:	Van Romero
Affiliation:	Sandia National Laboratories
Education:	PhD, Physics, State University of New York MS, Physics, New Mexico Tech BS, Physics, New Mexico Tech
Experience/ Technical Specialty:	Fifteen years. Environmental health physics and radiation protection, NEPA compliance, DOE order compliance, environmental impact testing, risk assessment, nuclear safety, health physics, radiation transport, and nuclear emergency response.
EIS Responsibility:	Radiation exposure analysis for marine transport

Name:	William B. Samuels
Affiliation:	Science Applications International Corporation
Education:	PhD, Biology, Fordham University MS, Marine Science, Long Island University BS, Biology & Geology, University of Rochester
Experience/ Technical Specialty:	Sixteen years. Geographic Information Systems, computer simulation and mathematical modeling, environmental database management systems.
EIS Responsibility:	Geographic Information Systems, environmental justice

Name:	Elizabeth C. Saris
Affiliation:	Science Applications International Corporation
Education:	BA, Political Science, George Washington University
Experience/ Technical Specialty:	Fifteen years. Energy and environmental policy analysis, public outreach, and technical writing.
EIS Responsibility:	EIS Summary, public hearings support

Name:	Patrick R. Schwab
Affiliation:	Science Applications International Corporation
Education:	PhD, Nuclear Engineering, University of Wisconsin MS, Nuclear Engineering, University of Wisconsin BS, Nuclear Engineering, Kansas State University
Experience/ Technical Specialty:	Eighteen years. Design criteria, technical safety surveys, foreign nuclear technology analysis, configuration studies, and spent fuel reprocessing.
EIS Responsibility:	Environmental and policy consequences, chemical separation technologies and impacts

Name:	Barry Smith
Affiliation:	Science Applications International Corporation
Education:	JD, George Washington University National Law Center BA, Political Science, Indiana University
Experience/ Technical Specialty:	Twenty-three years. NEPA compliance, environmental law, regulatory compliance, and waste management.
EIS Responsibility:	Environmental regulation/compliance

Name:	Jeremy L. Sprung
Affiliation:	Sandia National Laboratories
Education:	PhD, Physical-Organic Chemistry, UCLA BA, Chemistry, Yale University
Experience/ Technical Specialty:	Twenty-nine years. Photochemistry and air pollution, reactor accident consequences, reactor safety studies, and transportation risk assessment.
EIS Responsibility:	Port accident risk analysis

Name:	Donna J. Stucky
Affiliation:	Pacific Northwest Laboratories
Education:	MS, Agricultural Economics, Purdue University BA, Economics, Pacific Lutheran University
Experience/ Technical Specialty:	Two years. Economic research.
EIS Responsibility:	Environmental consequences

Name:	Robert Wayland
Affiliation:	Science Applications International Corporation
Education:	PhD, Atmospheric Science, North Carolina State University MS, Environmental Science, University of Virginia BA, Environmental Science, University of Virginia
Experience/ Technical Specialty:	Eleven years. Boundary-layer meteorology, atmospheric structure and composition, ocean-atmosphere interactions, atmospheric modeling.
EIS Responsibility:	Port meteorological data assessments, site nonradiological impact analyses

Name:	Timothy Wheeler
Affiliation:	Sandia National Laboratories
Education:	MS, Systems Engineering, University of Virginia BS, Mechanical Engineering, University of New Hampshire
Experience/ Technical Specialty:	Fourteen years. NEPA compliance, radioactive material transportation risk analysis, probabilistic risk assessment.
EIS Responsibility:	At-sea submerged cask risk assessment

Name:	John W. Williams
Affiliation:	Science Applications International Corporation
Education:	PhD, Physics, New Mexico State University MS, Physics, New Mexico State University BS, Mathematics, North Texas State University
Experience/ Technical Specialty:	Twenty years. NEPA compliance, electromagnetic models, air quality modeling, ionizing radiation impacts and safety.
EIS Responsibility:	Environmental justice, ports selection, quality control reviews

Name:	Steven E. Wujciak
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MBA, Anna Maria College BS, Business Administration, Anna Maria College
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name:	Maron D. Wylie
Affiliation:	U.S. Department of Transportation, Research & Special Projects Administration, Volpe National Transportation System Center
Education:	MS, Math and Computer Science, Worcester State College BS, Business Administration, University of Southern Mississippi
Experience/ Technical Specialty:	Fifteen years. Operations research, transportation analysis, emergency preparedness.
EIS Responsibility:	Ground transportation analysis

Name	Michael R. Zanotti
Affiliation:	Science Applications International Corporation
Education:	MPA, Administrative Management and Organization, Golden Gate University MPA, Health Services Administration, Golden Gate University BA, Behavioral Sciences, University of Maine AA, Criminal Justice, University of Maine
Experience/ Technical Specialty:	Fifteen years. Certified Emergency Manager (CEM), Emergency management, emergency response, fire response, hazardous materials response, facilities operation.
EIS Responsibility:	Emergency management and response

7. Agencies Consulted

The following agencies were consulted in the development of this Draft Environmental Impact Statement.

Federal Agencies

Arms Control and Disarmament Agency
Military Traffic Management Command
Military Ocean Terminal, Oakland (CA)
Military Ocean Terminal, Sunny Point (NC)
Naval Weapons Station, Concord (CA)
Naval Weapons Station, Charleston (SC)
Port Hueneme (CA) Naval Construction Battalion Center
U.S. Department of Defense
U.S. Department of Army
U.S. Coast Guard
U.S. Merchant Marine Academy
U.S. Fish and Wildlife Service

State Agencies

Alabama Department of Conservation and Natural Resources
Alabama Department of Environmental Management, Water Quality Division
Alabama Natural Heritage Program
Alabama State Docks, Mobile (AL)
California Fish & Game Heritage Program
California Regional Water Quality Control Board, San Francisco Bay Region
Delaware Department of Natural Resources and Environmental Control, Division of Water Resources
Delaware Natural Heritage Inventory
Florida Department of Environmental Regulation, Bureau of Surface Water Management
Florida Natural Areas Inventory
Fort Clinch State Park, Amelia Island, FL
Georgia Department of Natural Resources, Environmental Protection Division
Georgia Department of Natural Resources, Wildlife Resources Division
Georgia Ports Authority
John U. Lloyd Beach State Recreation Area, Port Everglades, FL
Louisiana Department of Environmental Quality
Louisiana Natural Heritage Program
Maryland Natural Heritage Program
Massachusetts Port Authority
Maryland Port Administration
Mississippi Department of Environmental Quality, Water Quality Division
Mississippi Natural Heritage Program
Mississippi State Port Authority at Gulfport
New Hampshire Port Authority
New Jersey Natural Heritage Program
North Carolina Department of Environment, Health, and Natural Resources, Division of Environmental Management
North Carolina Natural Heritage Program
North Carolina State Ports Authority
Oregon Natural Heritage Program
Pennsylvania Department of Environmental Resources, Water Quality Division
Pennsylvania Natural Diversity Inventory
Ports Authority of New York & New Jersey
South Carolina Department of Health and Environmental Control, Water Quality Division
South Carolina Heritage Trust
South Carolina State Ports Authority
South Jersey Port Corporation
Virginia Department of Conservation and Recreation, Division of Natural Heritage
Virginia Department of Environmental Quality, Water Division
Virginia Water Control Board
Virginia Port Authority
Washington Department of Wildlife

Local Agencies

Bridgeport Port Authority (CT)
Commissioners of Pilotage, Port of Charleston (SC)
Jacksonville Port Authority (FL)
Manatee County Port Authority (FL)
Penn Terminals (Port of Eddystone, PA)
Port Authority of Greater New Orleans (LA)
Port Everglades Authority (FL)
Port of Albany (NY)
Port of Alexandria (VA)
Port of Baton Rouge (LA)
Port of Beaumont (TX)
Port of Corpus Christi (TX)
Port of Fall River (MA)

Local Agencies (Continued)

Port of Fernandina (FL)
Port of Galveston (TX)
Port of Grays Harbor (WA)
Port of Houston Authority (TX)
Port of Hueneme (CA)
Port of Long Beach (CA)
Port of Longview (WA)
Port of Los Angeles (CA)
Port of Miami (FL)
Port of New Haven (CT)
Port of Oakland (CA)
Port of Palm Beach (FL)
Port of Port Arthur (TX)
Port of Portland (ME)
Port of Portland (OR)
Port of Portsmouth (NH)
Port of Richmond (CA)
Port of Richmond Commission (VA)
Port of San Francisco (CA)
Port of Seattle (WA)
Port of Tacoma (WA)
Port of Vancouver, U.S.A. (WA)
Port of Wilmington (DE)
Port of Wilmington (NC)
San Diego Unified Port District (CA)
Tampa Port Authority (FL)

Other

Australian Nuclear Science & Technology Organization (ANSTO)
Austrian Research Centre, Austria
Belgian Nuclear Research Centre
GKSS Research Center, Germany
Hahn-Meitner Institut Berlin, Germany
Interfaculty Reactor Institute, Delft University of Technology, The Netherlands
Joint Research Centre-Petten, Institute for Advanced Materials, The Netherlands
National Center for Scientific Research, "Demokritos," Greece
Paul Scherrer Institute, Switzerland
RISO National Laboratory, Denmark
Studsvik Nuclear AB, Sweden
United Kingdom Atomic Energy Authority, Thurso, Dounreay Caithness, Scotland

8. References

AAPA (American Association of Port Authorities), 1993, *The 1993 AAPA Directory, Seaports of the Western Hemisphere*, K-111 Directory Corporation, New York, NY, January.

Abbott, M. L., J. M. Brooks and K. L. Martin, 1990, *NPR Environmental Impact at the INEL: Air Quality, Cooling Tower, and Noise*, NPRD-90-059, EG&G Idaho, Inc., Idaho Falls, ID, November.

Ackerman, D. J., 1991, *Transmissivity of the Perched Aquifers at the INEL*, WRIR 91-4114, DOE/ID-22099, U.S. Geological Survey, Idaho Falls, ID.

Aiken County, 1991, *Functional Performance Standards for Non-residual Uses, Zoning and Development Standards Ordinance*, Chapter VI, Section 24-204 - Noise, Aiken, SC.

Analysas Corporation, 1994, *TRANSCOM Shipment and Analysis of Urgent Relief Training Statistics Facsimile*, Oak Ridge, TN, September 21.

Anderson, S. R., 1991, *Stratigraphy of the Unsaturated Zone and Uppermost Part of the Snake River Plain Aquifer at the Idaho Chemical Processing Plant and Test Reactor Area*, WRIR 91-4010, DOE/ID-22095, U.S. Geological Survey, Idaho Falls, ID, January.

ANSI (American National Standards Institute), 1975, Standard N305-1975, *Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant,* March.

Arnett, M. W., L. K. Karapatakis and A. R. Mamatey, 1993, *Savannah River Site Environmental Report for 1992*, WSRC-TR-93-075, Westinghouse Savannah River Company, Aiken, SC.

Arthur, W. J., J. W. Connelly, D. K. Halford and T. D. Reynolds, 1984, *Vertebrates of the Idaho National Engineering Laboratory*, DOE/ID-12099, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, ID, July.

Barraclough, J. T., B. D. Lewis and R. G. Jensen, 1981, *Hydrologic Conditions at the INEL, Idaho-Emphasis: 1974-1978*, WRI Open File Report 81-526, IDO-22060, U.S. Geological Survey, Idaho Falls, ID, April.

Beavers, J. E., W. E. Manrod and W. C. Stoddart, 1982, *Recommended Seismic Hazard Levels for the Oak Ridge, Tennessee; Paducah, Kentucky; Fernald, Ohio; and Portsmouth, Ohio, Department of Energy Reservations*, K/BD-1025/R1, Oak Ridge, TN, December.

Bendixsen, C. L., 1995, Lockheed Idaho Technologies Company, personal communication to A. Zeitoun, Science Applications International Corporation, October 18.

Bennett, C. M., 1990, *Streamflow Losses and Ground-Water Level Changes Along the Big Lost River at the INEL*, WRIR 90-4067, DOE/ID-22091, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, ID.

BNFL (British Nuclear Fuels Limited), 1994, *UK-Based Processing as a Spent Fuel Management Option for U.S. DOE*, Risley, England, May.

Bolt, B. A., 1978, *Earthquakes: A Primer*, W. H. Freeman & Co., San Francisco, CA.

Bowdle, K., 1994, U.S. Department of Energy, Environmental Restoration Division, personal communication with K. Laudkrohn, Haliburton NUS Corporation, Gaithersburg, MD, May 20.

Boyle, J. W., R. Blumbert, S. J. Cotter, G. S. Hill, C. R. Kerley, R. H. Ketelle, R. L. Kroodsma, D. W. Lee, R. C. Martin, R. D. Roop, D. N. Secora, W. P. Staub and R. E. Thoma, 1982, *Environmental Analysis of the Operation of Oak Ridge Laboratory (X-10 Site)*, ORNL-5870, Oak Ridge National Laboratory, Oak Ridge, TN, November.

Braun, J. B., S. J. Miller and B. L. Ringe, 1993, *Historically Significant Scientific and Technical Facilities at the INEL (Draft)*, EGG-CS-10699, EG&G Idaho, Inc., Idaho Falls, ID, March.

Breslin, S., 1993, Texas Natural Heritage Program, Texas Parks and Wildlife Department, personal communication to M. Shafer, Science Applications International Corporation, September 15.

Butz, T. W., 1984, *Resource Management Plan for DOE Oak Ridge Reservation, Volume 8, Appendix H: Geology*, ORNL-6026/V8, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, July.

Cartwright, P. (United Kingdom Atomic Energy Authority), 1995, personal communication to H. Chernoff, Science Applications International Corporation, March 16.

Cartwright, P. (United Kingdom Atomic Energy Authority), 1994, personal communication to A. Murray, Science Applications International Corporation, November 8.

Cholewa, A. F. and D. M. Henderson, 1984, *A Survey and Assessment of the Rare Vascular Plants of the INEL*, DOE/ID-12100, U.S. Department of Energy, Idaho Operations Office, Radiological and Environmental Sciences Laboratory, Idaho Falls, ID, July.

Clawson, K. L., G. E. Start and N. R. Ricks, 1989, *Climatography of the INEL*, second edition, DOE/ID-12118, National Oceanic and Atmospheric Administration, Idaho Falls, ID, December.

COGEMA, 1994, *Nuclear Materials Reprocessing and Recycling*, Paris, France, November.

Cottam, R., 1995, Lockheed Idaho Technology Company, personal communication to P. Schwab, Science Applications International Corporation, Germantown, MD, January 5.

CRC (CRC Press), 1991, *CRC Handbook of Chemistry and Physics*, 72nd Edition, 1991-1992, Boca Raton, FL.

Cummins, C. L., D. K. Martin and J. L. Todd, 1990, *Savannah River Site Environmental Report for 1989*, WSRC-IM-90-60, Westinghouse Savannah River Company, Aiken, SC.

Cunningham, M. and L. Pounds, 1991, *Resource Management Plan for the Oak Ridge Reservation, Volume 28, Wetlands on the Oak Ridge Reservation*, ORNL/NERP-5, Martin Marietta Energy Systems, Inc., Oak Ridge, TN, December.

Dahlke, H. J., D. A. Johnson, J. K. Rawlins, D. K. Searle, and G. W. Wachs, 1994, *Summary Engineering Description of Underwater Fuel Storage Facility for Foreign Research Reactor Spent Nuclear Fuel*, INEL-94/0014, Idaho Falls, ID, October.

Denney, R., 1995, Lockheed Idaho Technology Company, personal communication to A. Zeitoun, Science Applications International Corporation, Germantown, MD, January 25.

Department of the Navy, 1994, *Environmental Assessment, Construction of Magazines and Rail Lines, Naval Weapons Station, Charleston, South Carolina*, Public Works Officer, Naval Weapons Station, Charleston, SC, July.

Department of the Navy, 1990, *Environmental Assessment for Propulsion Training Facility, Naval Weapons Station Charleston, Berkeley County, South Carolina*, Southern Division, Naval Facility Engineering Command, Charleston, SC, July.

de Planque, E. G., 1994, *In Search of ... Background*, Remarks before the NRC Workshop on Site Characterization for Decommissioning, U.S. Nuclear Regulatory Commission, Rockville, MD, November 29.

DOC (U.S. Department of Commerce, Bureau of the Census), 1994, *Statistical Abstract of the United States, 114th Edition*, Washington, DC.

DOE (U.S. Department of Energy), 1995a, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, Savannah River Site, Aiken, SC, October.

DOE (U.S. Department of Energy), 1995b, *Draft Waste Management Programmatic Environmental Impact Statement for Managing Treatment, Storage, and Disposal of Radioactive and Hazardous Waste*, DOE/EIS-0200-D, Office of Environmental Management, Washington, DC, August.

DOE (U.S. Department of Energy), 1995c, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, Idaho Operations Office, Idaho Falls, ID, April.

DOE (U.S. Department of Energy), 1994a, *Integrated Data Base Report-1993: U.S. Spent Nuclear Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-0006, Rev. 10, Office of Civilian Radioactive Waste Management and Office of Environmental Management, Washington, DC, December.

DOE (U.S. Department of Energy), 1994b, *Final F-Canyon Plutonium Solutions Environmental Impact Statement*, DOE/EIS-0219, Savannah River Site, Aiken, SC, December.

DOE (U.S. Department of Energy), 1994c, *DOE Spent Nuclear Fuel Technology Integration Plan*, SNF-PP-FS-002, Office of Environmental Management, Washington, DC, December.

DOE (U.S. Department of Energy), 1994d, *Shipment Mobility and Accountability Collection (SMAC) Special Report*, Oak Ridge, TN, December 1.

DOE (U.S. Department of Energy), 1994e, *Final Supplemental Environmental Impact Statement, Defense Waste Processing Facility*, DOE/EIS-0082-S, Savannah River Site, Aiken, SC, November.

DOE (U.S. Department of Energy), 1994f, *A Disposition Strategy for Highly Enriched Aluminum-Based Fuel from Research and Test Reactors*, Savannah River Site, Aiken, SC, November 10.

DOE (U.S. Department of Energy), 1994g, *Financial Assistance to States and Tribes to Support Emergency Preparedness and Response and the Safe Transportation of Hazardous Shipments*, TTP No. RL-327001, Richland, WA, October.

DOE (U.S. Department of Energy), 1994h, *Implementation Plan for the Environmental Impact Statement on the Proposed Policy for the Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EIS-0218, Washington, D.C., October.

DOE (U.S. Department of Energy), 1994i, *Transportation External Coordination/Working Group (TEC/WG) List of Invited Organizations and Previous Participants*, Germantown, MD, September 28.

DOE (U.S. Department of Energy), 1994j, *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement*, DOE/EIS-0203-D, Idaho Operations Office, Idaho Falls, ID, June.

DOE (U.S. Department of Energy), 1994k, *Safety Analysis Report (200-S Area), Savannah River Site Defense Waste Processing Facility Operations*, DPSTSA-200-10, Supplement 20, Savannah River Site, Aiken, SC, June.

DOE (U.S. Department of Energy), 1994l, "Notice of Intent to Prepare a Programmatic Environmental Impact Statement for Storage and Disposition of Weapons-Usable Fissile Materials," *Federal Register*, Volume 59, No. 118, June 21.

DOE (U.S. Department of Energy), 1994m, *Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EA-0912, Washington, DC, April.

DOE (U.S. Department of Energy), 1994n, *Report on the Emergency Response Training and Equipment Activities through Fiscal Year 1993 for the Transportation of Transuranic Waste to the Waste Isolation Pilot Plant*, DOE/WIPP 93-061, Carlsbad, NM, April.

DOE (U.S. Department of Energy), 1994o, *Waste Acceptance System Requirements Document*, Revision 01, DOE/RW-0351P Rev. 1, February.

DOE (U.S. Department of Energy), 1993a, *Annual Site Environmental Report - 1992, Volume 1 and Volume 2*, DOE/NV/10630-66, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1993b, *1993 NTS Technical Site Information Package - Volumes 1 and 2*, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1993c, *Tornado Special Study*, ES/ESH-35, June 14.

DOE (U.S. Department of Energy), 1993d, *Nonnuclear Consolidation Environmental Assessment, Volume 1, Nuclear Weapons Complex Reconfiguration Program*, DOE/EA-0792, Office of Defense Programs, Washington, DC, June.

DOE (U.S. Department of Energy), 1993e, *Transporting Radioactive Materials - Q&A to Your Questions*, DOE\EM-0097, Washington, DC, April.

DOE (U.S. Department of Energy), 1992a, *DOE Standard: Hazard Categorization Accident Analysis Technique for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Report*, DOE-STD-1027-92, December.

DOE (U.S. Department of Energy), 1992b, *Integrated Data Base for 1992: U.S. Spent Fuel and Radioactive Waste Inventories, Projections, and Characteristics*, DOE/RW-006, Rev. 8, Office of Civilian Radioactive Waste Management, Washington, DC, October.

DOE (U.S. Department of Energy), 1992c, *U.S. Department of Energy Nevada Field Office Annual Site Environmental Report - 1991*, DOE/NV/10630-33, Nevada Operations Office, Las Vegas, NV, September.

DOE (U.S. Department of Energy), 1991a, *Draft Environmental Impact Statement for the Siting, Construction and Operation of New Production Reactor Capacity*, DOE/EIS-0144D, Office of New Production Reactors, Washington, DC, April.

DOE (U.S. Department of Energy), 1991b, *U.S. Proposal for the Nuclear Weapons Complex Reconfiguration Site*, Savannah River Field Office, Aiken, SC.

DOE (U.S. Department of Energy), 1991c, *Planning and Preparedness for Operational Emergencies*, DOE Order 5500.3A, April 30.

DOE (U.S. Department of Energy), 1990a, *Final Environmental Impact Statement, Continued Operation of K-, L-, and P-, Reactors, Savannah River Plant, Aiken, South Carolina*, DOE/EIS-0147, Savannah River Operations Office, Aiken, SC, December.

DOE (U.S. Department of Energy), 1990b, *Review of the Emergency Management System and Its Preparedness to Respond to Transportation Incidents,* TITF-91-001, November.

DOE (U.S. Department of Energy), 1990c, *Radiation Protection of the Public and the Environment*, DOE Order 5400.5, February 8.

DOE (U.S. Department of Energy), 1989, *Shipment of Radioactive Material by the U.S. Department of Energy,* DOE/DP-0065, Washington, DC.

DOE (U.S. Department of Energy), 1988a, *Site Characterization Plan, Yucca Mountain Site, Nevada Research and Development Area, Nevada*, DOE/RW-0199, Office of Civilian Radioactive Waste Management, December.

DOE (U.S. Department of Energy), 1988b, *Status of Flora and Fauna on the Nevada Test Site*, DOE/NV/10630-29, Nevada Operations Office, Las Vegas, NV.

DOE (U.S. Department of Energy), 1988c, *Consultant Draft: Site Characterization Plan, Reference Repository Location, Hanford Site, Washington*, DOE/RW-0164, Washington, DC, January.

DOE (U.S. Department of Energy), 1987, *Final Environmental Impact Statement, Waste Management Activities for Groundwater Protection, Savannah River Plant, Aiken, South Carolina*, DOE/EIS-0120, Savannah River Operations Office, Aiken, SC, December.

DOE (U.S. Department of Energy), 1986a, *Environmental Assessment, Reference Repository Location, Hanford Site, Washington,* DOE-RW-0070, Office of Civilian Radioactive Waste Management, Washington, DC, May.

DOE (U.S. Department of Energy), 1986b, *Environmental Assessment, Yucca Mountain Site, Nevada Research and Development Area, Nevada*, DOE/RW-0073, Office of Civilian Radioactive Waste Management, Washington, DC, May.

DOE (U.S. Department of Energy), 1982, *Geology of the Nevada Test Site and Nearby Areas, Southern Nevada*, SAND82-2207, Sandia National Laboratories, October.

DOE (U.S. Department of Energy), 1980, *Final Environmental Impact Statement, United States Spent Fuel Policy*, DOE/EIS-0015, Washington, DC, May.

Dukes, E. K., 1984, *The Savannah River Plant Environment*, DP-1642, E.I. du Pont de Nemours and Company, Inc., Aiken, SC, March.

Dupont, M.E., 1996, Westinghouse Savannah River Company, personal communication to I. Zeitoun, Science Applications Internatinal Corporation, January 10.

Eckel, E. B., 1968, *The Geologic Society of America, Inc. Memoir 110, Nevada Test Site*, The Geologic Society of America, Boulder, CO.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1993, *Waste Management Facilities Cast Information Estimating Data for Spent Nuclear Fuel*, EGG-WM-10708, March.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1984, *INEL Environmental Characterization Report*, EGG-NPR-6688, Idaho Falls, ID, September.

EPA (U.S. Environmental Protection Agency), 1993a, *Attainment Status Designations, South Carolina*, 40 CFR 81(Revised as of July 1, 1992), Office of the Federal Register, Washington, DC.

EPA (U.S. Environmental Protection Agency), 1993b, *Prevention of Significant Deterioration of Air Quality*, 40 CFR 52.21 (Revised as of July 1, 1992), Office of the Federal Register, Washington, DC.

EPA (U.S. Environmental Protection Agency), 1991a, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, EPA 400-R-92-001, Office of Radiation Programs, Washington, DC, October (Reprinted May 1992).

EPA (U.S. Environmental Protection Agency), 1991b, "Sole Source Designation of the Eastern Snake River Plain Aquifer, Southern Idaho, Final Determination," *Federal Register*, Volume 56, No. 194, October.

ERDA (U.S. Energy Research and Development Administration), 1976, *Ecology of the Nevada Test Site: A Narrative Summary and Annotated Bibliography*, Las Vegas, NV, May.

FEMA (Federal Emergency Management Agency), 1994, *Cooperative Contract Agreement Funding Documents for Fiscal Years 1982-1993,* October.

FEMA (Federal Emergency Management Agency), 1985, "Federal Radiological Emergency Response Plan," *Federal Register*, Volume 50, No. 217, November 8.

FHI (Frederick R. Harris, Inc.), 1993, *Kings Bay Naval Station Submarine Base; Assessment for Receipt of FRR SNF*, Iselin, NJ, November 1.

FICON (Federal Interagency Committee on Noise), 1992, *Federal Agency Review of Selected Airport Noise Analysis Issues*, U.S. Department of Transportation, Federal Aviation Administration, Washington, DC.

Fitzner, R. E. and R. H. Gray, 1991, "The Status, Distribution, and Ecology of Wildlife on the U.S. DOE Hanford Site: A Historical Overview of Research Activities," *Environmental Monitoring and Assessment*, Volume 18, pp. 173-202.

Friedrich, H., 1969, *Marine Biology: An Introduction to Its Problems and Results*, University of Washington Press, Seattle, WA.

FWS (U.S. Fish and Wildlife Service), 1992, *Threatened and Endangered Species of Texas*, Texas State Office, Austin, TX, August.

FWS (U.S. Fish and Wildlife Service), 1981a, *Pacific Coast Ecological Inventory Map*: 38120-A1-EI-250, Sacramento, CA.

FWS (U.S. Fish and Wildlife Service), 1981b, *Pacific Coast Ecological Inventory Map*: 38122-A1-EI-250, Santa Rosa, CA.

FWS (U.S. Fish and Wildlife Service), 1981c, Pacific Coast Ecological Inventory, 45122-A1-E1-250, Vancouver, Washington, Oregon.

FWS (U.S. Fish and Wildlife Service), 1980a, *James Island Atlantic Coast Ecological Inventory*, 32078-A1-EI-250, James Island, SC.

FWS (U.S. Fish and Wildlife Service), 1980b, *Atlantic Coast Ecological Inventory*, 30080-A1-EI-250, Jacksonville, FL-GA.

FWS (U.S. Fish and Wildlife Service), 1980c, *Savannah, Atlantic Coast Ecological Inventory Map*, 32080-A1-EI-250, Savannah, GA.

Gaia Northwest, Inc., 1988, *Drinking Water Consumption and Alternative Sources for the Eastern Snake River Plain, Idaho*, November.

Gaines, E., 1994, Oregon National Heritage Program, personal communication to D. Scott, Science Applications International Corporation, April 12.

GAO (U.S. General Accounting Office), 1994, *Foreign Countries' Approaches to High-Level Waste Storage and Disposal*, GAO/RCED-94-172, Washington, DC, August.

Garabedian, S. P., 1989, *Hydrology and Digital Simulation of the Regional Aquifer System, Eastern Snake River Plain, Idaho*, USGS Open File Report 87-237, U.S. Geological Survey.

Garabedian, S. P., 1986, *Application of a Parameter Estimation Technique to Modeling the Regional Aquifer Underlying the Eastern Snake River Plain, Idaho*, Water Supply Paper 2278, U.S. Geological Survey.

Gilbert, H. K. and B. L. Ringe, 1993, *Inventory of Known Historical Cultural Resources on the INEL and Preliminary Analysis of Historic Sensitivity (Draft)*, EGG-CS-10707, EG&G Idaho Inc., Idaho Falls, ID, March.

Golder Associates, 1993, *Assessment of Trends in Groundwater Quality at the Idaho National Engineering Laboratory*, Report No. 933-1151, Idaho Falls, ID, October 29.

Gonzalez, A., J. Pierce and D. Stenberg, 1986, *Target Hardness Comparison to IAEA Unyielding Target*, IAEA-SM-114, SAND86-0052c, Sandia National Laboratories, Albuquerque, NM.

Gordon-Hagerty, L. E., 1993, "United States Department of Energy Radiological Emergency Response Programme - A National Capability," *Radiation Protection Dosimetry*, Volume 50, Nos. 2-4, pp. 153-156.

Graham, M. J., M. D. Hall, S. R. Strait and W. R. Brown, 1981, *Hydrology of the Separations Area*, RHO-ST-42, Rockwell Hanford Operations, Richland, WA, July.

Grainger, A. R., 1995, *Radiological Emissions Calculations for Savannah River Site (SRS) Processing Activities*, Memorandum to Charles Head from Savannah River Operations Office, July 31, 1995.

Greene, G. S., 1975, *Prehistoric Utilization of the Channeled Scablands of Eastern Washington*, Ph.D Dissertation, Washington State University, Department of Anthropology, Pullman, WA.

Greger, P. D., E. M. Romney, n.d.a., *Wildlife Utilization of National Springs and Man-Made Sources at the Nevada Test Site (Draft)*, UCLA Laboratory of Biomedical and Environmental Sciences.

Hackett, W. R. and R. P. Smith, 1992, "Quaternary Volcanism, Tectonics, and Sedimentation in the Idaho National Engineering Laboratory Area," in *Field Guide to Geologic Excursions in Utah and Adjacent Areas of Nevada, Idaho, Wyoming*, by J. R. Wilson, Geological Society of America, Rocky Mountain Section Guidebook, Utah Geological Survey Miscellaneous Publication 92-3, pp. 1-18.

Hajek, B. F., 1966, *Soil Survey: Hanford Project in Benton County, Washington*, BNWL-243, Pacific Northwest Laboratory, Richland, WA, April.

Hatcher, Jr., R. D., P. J. Lemiszki, R. B. Dreier, R. H. Ketelle, R. R. Lee, D. A. Lietzke, W. M. McMaster, J. L. Foreman and S. Y. Lee, 1992, *Status Report on the Geology of the Oak Ridge Reservation*, Oak Ridge National Laboratory, Environmental Services Division Publication No. 3860, ORNL/TM-12074, October.

Hessler, R. R. and H. L. Sanders, 1967, "Faunal Diversity in the Deep Sea," *Deep Sea Resources*, Volume 14, pp. 65-78.

HNUS (Haliburton NUS Environmental Corporation), 1993, *Forecasts Generated From the Regional Economic Models, Economic-Demographic Forecasting and Simulation Model*, Savannah River Center, Aiken, SC.

HNUS (Haliburton NUS Environmental Corporation), 1990, *Sound Level Characterization of the Savannah River Site*, NUS-5251, Savannah River Center, Aiken, SC, August.

Hoff, D. L., R. G. Mitchell, G. C. Bowman and R. Moore, 1990, *The Idaho National Engineering Laboratory Site Report for Calendar Year 1989*, DOE/ID-12082(89), Idaho Operations Office, June.

Holm, J. A., 1994, "Transportation External Coordination/Working Group," U.S. Department of Energy Transportation Management Workshop, Rockville, MD, August 9.

IAEA (International Atomic Energy Agency), 1992, *Effects of Ionizing Radiation on Plants and Animals at Levels Implied by Current Radiation Protection Standards*, Technical Report Series No. 332, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1990, *Regulations for the Safe Transport of Radioactive Material*, 1985 Edition as Amended 1990, Safety Series No. 6, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1988, *Assessing the Impact of Deep Sea Disposal of Low Level Radioactive Waste on Living Marine Resources*, Technical Reports Series No. 288, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1976, *Effects of Ionizing Radiation on Aquatic Organizations and Ecosystems*, Technical Report Series No. 172, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection*, Annals of the ICRP, Publication 60, Volume 21, No. 1-3, Pergamon Press, New York, NY, November.

IMO (International Maritime Organization), 1994, *International Maritime Dangerous Goods Code*, London, UK.

IMO (International Maritime Organization), 1993, Resolution A, 748 (18), *Code for the Safe Carriage of Irradiated Nuclear Fuel, Plutonium, and High-Level Radioactive Wastes in Casks on Board Ships*, London, UK, adopted November 4.

IMO (International Maritime Organization), 1992, *SOLAS - Consolidated Text of the International Convention for the Safety of Life at Sea*, London, UK.

IPA (Information Please Almanac), 1993, *The 1993 Information Please Almanac, Atlas, and Yearbook*, 46th Edition, Houghton Mifflin Company, Boston, MA.

Irving, J. S., 1993, *Environmental Resource Document for the Idaho National Engineering Laboratory*, Volume 1, EGG-WMO-10279, July.

Jane's (Jane's Information Group, Inc.), 1992, *Jane's Containerization Directory*, 24th Edition, 1992-1993, Alexandria, VA.

Jefferson, R. M. and H. R. Yoshimura, 1978, *Crash Testing of Nuclear Fuel Shipping Containers*, SAND77-1462, Sandia National Laboratories, Albuquerque, NM, February.

Johnson, P. E., D. S. Joy, D. B. Clark and J. M. Jacobi, 1993a, *HIGHWAY 3.1, An Enhanced Highway Routing Model: Program Description, Methodology, and Revised User's Manual*, ORNL/TM-12124, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Johnson, P. E., D. S. Joy, D. B. Clark and J. M. Jacobi, 1993b, *INTERLINE 5.0, An Expanded Railroad Routing Model: Program Description, Methodology, and Revised User's Manual*, ORNL/TM-12090, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Jones, G. P., R. W. Allott, V. J. Dennis, J. P. Fowler, B. Hughes, D. Lomas and J. Wilkinson, 1994, *Radioactive waste discharges from UKAEA establishments during 1993 and associated monitoring results*, UKAEA/CSD R012, United Kingdom Atomic Energy Authority, August.

Jow, H. N., J. L. Sprung, J. A. Rollstin, L. T. Ritchie and D. I. Chanin, 1990, *MELCOR Accident Consequences Code System (MACCS), Model Description*, NUREG/CR-4691, SAND86-1562, Volume 2, Sandia National Laboratories, Albuquerque, NM, February.

Kennett, J. P., 1982, *Marine Geology*, Prentice-Hall, Englewood Cliffs, NJ.

Ketelle, R. H. and D. D. Huff, 1984, *Site Characterization of the West Chestnut Ridge Site*, ORNL/TM-9229, Oak Ridge National Laboratory, Oak Ridge, TN.

King, A. L., D. J. Awl and C. A. Gabrielsen, 1994, *Environmentally Sensitive Areas Survey Program Threatened and Endangered Species Status Report*, Health Resources Division, Oak Ridge National Laboratory, Environmental Restoration Program (ES/ER/TM-130), September.

Knox, J. N. and R. R. Sharitz, 1990, *Endangered, Threatened, and Rare Vascular Flora of the Savannah River Site*, Savannah River Site National Environmental Research Park Publication, SRO-NERP-20, Aiken, SC.

Koch, H. F., 1984, *"Sinkholes: Their Geology, Engineering and Environmental Impact,"* Proceedings of First Multidisciplinary Conference on Sinkholes, Orlando, FL, October.

Kroodsma, R. L., 1987, *Resource Management Plan for the Oak Ridge Reservation: Volume 24, Threatened and Endangered Animal Species*, ORNL/ESH-1/V24, Oak Ridge National Laboratory, Oak Ridge, TN, January.

Kuntz, M. A., 1992, "A Model-based Perspective of Basaltic Volcanism, Eastern Snake River Plain, Idaho," in *Regional Geology of Eastern Idaho and Western Wyoming*, by P. K. Link, M. A. Kuntz and L. B. Platt, editors, Geological Society of America, Memoir 179, pp. 289-304.

Kurkoski, D., 1994, U.S. Army Corps of Engineers, personal communication to D. Scott, Science Applications International Corporation, March 24.

Lee, R. R. and R. H. Ketelle, 1987, *Stratigraphic Influence on Deep Groundwater Flow in the Knox Group Copper Ridge Dolomite on the West Chestnut Ridge Site*, ORNL/TM-10479, Oak Ridge National Laboratory, Oak Ridge, TN, October.

Leenheer, J. A. and J. C. Bagby, 1982, *Organic Solutes in Groundwater at the Idaho Engineering Laboratory*, Water Resources Investigation Report 82-15, U.S. Geological Survey, ID022061, Idaho Falls, ID, March.

Leonard, P. R., 1993, *Estimated Radiological Doses Resulting from Airborne Radionuclides Released by Facilities at the INEL*, EGG-WTD-100676, EG&G Idaho, Inc., Idaho Falls, ID, July.

Leventhal, P. L. and E. S. Lyman, 1995, Nuclear Control Institute, personal communication to H. O'Leary, U.S. Department of Energy, December 20.

Lewis, B., 1995, Naval Weapons Station Charleston, personal communication to G. DeMoss, Science Applications International Corporation, October 26.

Liszewski, M. J. and L. J. Mann, 1992, *Purgeable Organic Compounds in Groundwater at the Idaho National Engineering Laboratory*, DOE/ID 22104, Idaho Falls, ID, July.

Lloyd's (Lloyd's of London Maritime Information and Consultancy Services), 1991, *1991 Casualty File Manual,* London, England, November 18.

Mann, L. J. and L. L. Knobel, 1987, *Purgeable Organic Compounds at the Idaho National Engineering Laboratory*, Open File Report 766, U.S. Geological Survey, DOE/ID 22074, Idaho Falls, ID, December.

Masson, H., S. Runge, S. E. Ion and N. L. Griffin, 1994, "Waste Management Issues: Reprocessing or Direct Disposal," *Volume 1, RECOD '94*, Proceedings of Fourth International Conference on Nuclear Fuel Reprocessing and Waste Management, April.

Matos, J. E., 1994, *Foreign Research Reactor Irradiated Nuclear Fuel Inventories Containing HEU and LEU of U.S. Origin*, Argonne National Laboratory, ANL/RERTR/TM-22, December.

McBee, A., 1994, South Carolina Heritage Trust, South Carolina Wildlife & Marine Resources Department, personal communication to D. Scott, Science Applications International Corporation, April 5.

McKibben, J. M., T. H. Gould, W. R. McDonell, and W. E. Bickford, 1994, *A Disposition Strategy for Highly Enriched Aluminum-Based Fuel from Research and Test Reactors,* Westinghouse Savannah River Company, Aiken, SC, November.

McKinney, J. D., 1985, *Big Lost River 1983-1984 Flood Threat*, PPD-FPB-002, EG&G, Idaho Falls, ID, July.

MMES (Martin Marietta Energy Systems), 1994, *White Paper Summary Report: Spent Nuclear Fuels Management Complex Siting Study*, Oak Ridge, TN, March.

MMES (Martin Marietta Energy Systems), 1993a, *Oak Ridge Reservation Technical Site Information Draft,* ES/EN/SFP-23, Site and Facilities Planning Department, Oak Ridge, TN, September.

MMES (Martin Marietta Energy Systems), 1993b, *Oak Ridge Reservation Environmental Report for 1992,* ES/ESH-31, Oak Ridge, TN, June.

MMES (Martin Marietta Energy Systems), 1992, Oak Ridge Reservation Environmental Report for 1991, ES/ESH-22, October.

MMES (Martin Marietta Energy Systems), 1991, *Oak Ridge Reservation Site Development and Facilities Utilization Plan, 1990 Update,* DOE/OR-885(91), Oak Ridge, TN, June.

MMES (Martin Marietta Energy Systems), 1989, *Oak Ridge Reservation Site Development and Facilities Utilization Plan*, DOE/OR-885/R1, Oak Ridge, TN, June.

Murray, S., 1994, Florida Natural Areas Inventory, personal communication to D. Scott, Science Applications International Corporation, April 27.

NCRP (National Council on Radiation Protection and Measurements), 1991, *Effects of Ionizing Radiation on Aquatic Organisms,* NCRP Report No. 109, Bethesda, MD, March.

NEA (Nuclear Energy Agency), 1988, *Feasibility of Disposal of High-Level Radioactive Waste into the Seabed*, Office of Economic Cooperation and Development.

Neuhauser, K. S. and F. L. Kanipe, 1993, *RADTRAN 4, Volume II: Technical Manual*, SAND 89-2370, Sandia National Laboratories, Albuquerque, NM, August.

NNPP (Naval Nuclear Propulsion Program), 1993, *Occupational Radiation Exposure for U.S. Naval Nuclear Plants and Their Support Facilities*, Report NT-93-2, Washington, DC, February.

NOAA (National Oceanic and Atmospheric Administration), 1993a, *Local Climatological Data: Annual Summary with Comparative Data, Galveston, Texas*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993b, *Local Climatological Data: Annual Summary with Comparative Data, Los Angeles, California*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992a, *Local Climatological Data Annual Summary with Comparative Data - Charleston, South Carolina*, National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992b, *Local Climatological Data Annual Summary with Comparative Data - Jacksonville, Florida,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992c, *Local Climatological Data: Annual Summary with Comparative Data, Norfolk, Virginia,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992d, *Local Climatological Data Annual Summary with Comparative Data - Portland, Oregon,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992e, *Local Climatological Data Annual Summary with Comparative Data - Savannah, Georgia,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992f, *Local Climatological Data Annual Summary with Comparative Data - Seattle, Washington,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992g, *Local Climatological Data Annual Summary with Comparative Data - Wilmington, North Carolina,* National Climatic Data Center, Asheville, NC.

NRC (U.S. Nuclear Regulatory Commission), 1995, *Nuclear Regulatory Commission, 10 CFR Part 71, Compatibility with International Atomic Energy Agency (IAEA),* Federal Register, Volume 60, No. 188, September 28.

NRC (U.S. Nuclear Regulatory Commission), 1993, *Public Information Circular for Shipments of Irradiated Nuclear Fuel,* NUREG-0725, Revision 9, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 1991, *Standards for Protection Against Radiation* (Revised), 10 CFR Parts 2, 19, 20, 30, 31, 32, 34, 35, 39, 40, 50, 61, and 70, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Transporting Spent Fuel Protection Provided Against Severe Highway and Railroad Accidents,* NUREG/BR-0111, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,* NUREG-0170, Volumes I and II, Office of Standards Development, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1975, *Spent Fuel Storage Facility Design Basis,* Regulatory Guide 1.13, Revision 1, December.

ORNL (Oak Ridge National Laboratory), 1988, *Data Package for the Low-Level Waste Disposal Development and Demonstration Program, Environmental Impact Statement,* ORNL/RM-10939, Volume 1 and Volume 2, Oak Ridge, TN, September.

Orr, B. R. and L. D. Cecil, 1991, *Hydrologic Conditions and Distribution of Selected Chemical Constituents in Water, Snake River Plain Aquifer, Idaho National Engineering Laboratory, 1986-1988,* Water Resource Report 91-4047, U.S. Geological Summary, Idaho Falls, ID, March.

Parr, P. D. and J. W. Evans, 1992, *Draft Resource Management Plan for the Oak Ridge Reservation, Wildlife Management Plan,* ORNL/NERP-6, Oak Ridge National Laboratory, Oak Ridge, TN.

Parr, P. D. and L. R. Pounds, 1987, *Resource Management Plan for the Oak Ridge Reservation,* ORNL/ESH-1/V23, Oak Ridge National Laboratory, Oak Ridge, TN.

PIC (Planning Information Corporation), 1992, *A Socioeconomic Assessment of the Proposed Caliente Rail Spur, Nye County (Draft)*, Nevada, December.

Pickard, G. L. and W. J. Emery, 1982 *Descriptive Physical Oceanography*, 4th Edition, Pergamon Press.

Pittman, J. R., R. G. Jensen and P. R. Fischer, 1988, *Hydrologic Conditions at the INEL, 1982-1985*, U.S. Geological Survey Water Resources Investigation Report 89-4008, DOE/ID-22078, December.

Pounds, L. R., P. D. Parr and M. G. Ryon, 1993, *Resource Management Plan for the Oak Ridge Reservation, Volume 30: Oak Ridge National Environmental Research Park Natural Areas and Reference Areas - Oak Ridge Reservation Environmentally Sensitive Sites Containing Special Plants, Animals, and Communities*, ORNL-NERP-8, Oak Ridge National Laboratory, Oak Ridge, TN, August.

POW (Port of Wilmington), 1994, Port of Wilmington, DE, Brochure and Fact Sheet.

REAC/TS (Radiation Emergency Assistance Center/Training Site), n.d.a., *The Radiation Emergency Assistance Center/Training Site and World Health Organization Collaborating Center for Radiation Emergency Assistance,* Oak Ridge, TN.

Reynolds, T. D., 1993, Environmental Science and Research Foundation, U.S. Department of Energy, Idaho Operations Office, personal communication with T. Doerr, Science Applications International Corporation, September 8.

Rice, A. L., 1978, "Radioactive Waste Disposal and Deep Sea Biology," *Oceanol,* Acta 1, pp. 483-491.

Rice, D. G., 1968, *Archaeological Reconnaissance: Ben Franklin Reservoir Area, 1968*, Washington State University, Laboratory of Anthropology, Pullman, WA.

Rickard, W. H. and D. G. Watson, 1985, "Four Decades of Environmental Change and Their Influences Upon Native Wildlife and Fish on the Mid-Columbia River, Washington, U.S.A.," *Environmental Conservation*, Volume 12, No. 3, pp. 241-248.

Robertson, J. B., R. Schoen and J. T. Barraclough, 1974, *The Influence of Liquid Waste Disposal on the Geochemistry of Water at the National Reactor Testing Station, Idaho: 1952-1970*, IDO-22053, U.S. Geological Survey, February.

Rogers, L. E. and W. H. Rickard, 1977, *Ecology of the 200 Area Plateau Waste Management Environs: A Status Report,* PNL-2253, Pacific Northwest Laboratory, Richland, WA, October.

Rogoff, C., 1994, U.S. Department of Transportation, Research & Special Projects Administration, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, October 12.

Rohay, A. C., 1987, *Earthquake Focal Mechanisms, Recurrence Rates and Deformation in the Columbia River Basalts*, RHO-BW-SA-666P, Rockwell Hanford Operations, Richland, WA.

Rosensteel, B., 1994, *Wetlands and Rare Plant Information*, Internal Correspondence, Oak Ridge National Laboratory, Oak Ridge, TN, March 21.

RSN (Raytheon Services Nevada), 1993, *Summary of Natural Resources that Potentially Influence Human Intrusion at the Area 5 Radioactive Waste Management Site, DOE/Nevada Test Site, Nye County, Nevada*, Las Vegas, NV, August.

Sackschewsky, M. R., D. S. Landeen, G. I. Baird, W. H. Rickard and J. L. Downs, 1992, *Vascular Plants of the Hanford Site*, WHC-EP-0554, Westinghouse Hanford Company, Richland, WA, July.

Schalles, J. F., R. R. Sharitz, J. W. Gibbons, G. J. Leversee and J. N. Knox, 1989, *Carolina Bays of the Savannah River Plant*, National Environmental Research Park Publications, Savannah River Ecology Laboratory, SRO-NERP-18, Aiken, SC.

Schmitt, D. N., 1993, *Savannah and Ogeechee River Creek Surveys*, Georgia Department of Natural Resources, Game and Fish Division.

Schultz, T. J., 1978, "Synthesis of Social Surveys on Noise Annoyance," *Journal of the Acoustical Society of America*, Volume 64, No. 2, pp. 377-405, August.

Scott, W. E., 1982, *Surficial Geologic Map of the Eastern Snake River Plain and Adjacent Areas, Idaho and Wyoming*, U.S. Geological Survey, Miscellaneous Investigation Map I-1372.

Shedrow, C. B., 1993, *SRS Affected Environment*, WSRC-TR-93-667, Westinghouse Savannah River Company, Aiken, SC, October.

Shields, J. D., N. D. Woody, A. S. Dicks, G. J. Hollod, J. Schalles and G. J. Leversee, 1982, *Locations and Areas of Ponds and Carolina Bays at the Savannah River Plant*, DP-1525 (Revision 1), E.I. du Pont de Nemours and Company, Inc., Savannah River Laboratory, Aiken, SC.

Shirai, E., M. Adachi, T. Kodaira and H. Sato, 1991, "Dry Storage Facility of JRR-3 Spent Fuels", Third International Conference on Nuclear Fuel Processing and Waste Management, *RECOD '91*, Sendai, Japan, April 14-18, 1991.

Silver, J. M., 1993, Australian Nuclear Science and Technology Organization, Facsimile to J. Edlow, Edlow International Company, April 8.

Slovic, P., 1986, "Informing and Educating the Public About Risk," *Risk Analysis*, Volume 6, No. 4, pp. 403-415.

Smith, R. B. and W. J Arabasz, 1991, "Seismicity of the Intermountain Seismic Belt", *in* Slemmons, D. B., E. R. Engdahl, M. D. Zoback, and D. D. Blackwell, eds., *Neotectonics of North America:* Boulder, CO, Geological Society of America, Decade Map, Volume 1.

Solomon, D. K., G. K. Moore, L. E. Toran, R. B. Drier and W. M. McMaster, 1992, *Status Report: A Hydrologic Framework for the Oak Ridge Reservation*, Oak Ridge National Laboratory, Environmental Services Division Publication No. 3815, ORNL/TM-12026, Oak Ridge, TN, May.

SSEB (Southern States Energy Board), 1994, *Transportation Working Group on the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel Meeting Summary and Final Attendance List,* May 25-26.

State of North Carolina, 1994, Letter from R. M. Fry, Deputy Director of the Department of Environmental's Health and Natural Resources, to Ms. T. Vick, Blue Ridge Environmental Defense League, Glendale Springs, NC, October 10.

Stephenson, D. E., 1988, *August 1988 Savannah River Plant Earthquake*, DPST-88-841, E.I. du Pont de Nemours and Company, Inc., Savannah River Laboratory, Aiken, SC, September.

Stone, W. A., J. M. Thorp, O. P. Gifford and D. J. Hoitink, 1983, *Climatological Summary for the Hanford Area*, PNL-4622, Pacific Northwest Laboratory, Richland, WA, June.

Tallman, A. M., K. R. Fecht, M. C. Maratt and G. V. Last, 1979, *Geology of the Separation Areas Hanford Site, South-Central Washington*, RHO-ST-23, Rockwell Hanford Operations, Richland, WA, June.

Taylor, W., 1995, U.S. Department of Energy, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, January 20.

Taylor, W. and J. Hauptman, 1994, U.S. Department of Energy, personal communication to M. Zanotti, Science Applications International Corporation, Germantown, MD, September 28.

TDOT (Tennessee Department of Transportation), 1991, Tennessee City and County Maps Showing 1990 Average Daily Traffic, Bureau of Planning and Development, March 25.

TPWD (Texas Parks and Wildlife Department), 1989a, *Trends in Texas Commercial Fishery Landings,* 1977-1988, Management Data Series No. 7, Fisheries Division.

TPWD (Texas Parks and Wildlife Department), 1989b, *Trends in Finfish Landings by Sport-boat Fishermen in Texas Marine Waters,* May 1974-1988, Management Data Series No. 8, Fisheries Division.

U.S. Army (U.S. Army Corps of Engineers), 1991, *Environmental Impact Study, Savannah Harbor, Georgia Comprehensive Study*, July.

U.S. Federal Register, 1979, *Executive Order 12114 - Environmental Effects Abroad of Major Federal Actions*, 44 FR 1957, 3 CFR, 1979 Comp. p. 356, Washington, DC, January 4.

UBC (Uniform Building Code), *Uniform Building Code*, 1991, International Conference of Building Officials, Whittier, CA.

UKAEA (United Kingdom Atomic Energy Authority), 1994, *Capability of Dounreay Fuel Plants*, Government Division, Nuclear Sites Operations, Scotland, May.

USAF (U.S. Air Force), U.S. Navy and U.S. Department of Interior, 1991, *Special Nevada Report*, DE-AC08-88NV10715, Nellis Air Force Base, NV, September.

USDA (U.S. Department of Agriculture), 1992, *Savannah River Site, Wildlife, Fisheries, and Botany Operations Plan*, Savannah River Forest Station, New Ellenton, SC, December.

USMMA (U.S. Merchant Marine Academy) 1994, *Draft Report on the Workshop on Port Selection Criteria for Shipments of Foreign Research Reactor Spent Nuclear Fuel, Kings Point, NY,* Sandia National Laboratories, Albuquerque, NM, February 28.

WDW (Washington Department of Wildlife), 1994, *Priority Habitat and Species and Map*, Seattle South, No. 4712253, April.

Werner, F. T., 1994, U.S. Fish and Wildlife Service, Division of Ecological Services, personal communication to D. Scott, Science Applications International Corporation, November 2.

Wichmann, T., 1995, U.S. Department of Energy, Idaho Operations Office, personal communication to U.S. Department of Energy and contractors regarding Spent Nuclear Fuel Inventory Data, February 1.

Wilmot, E. L., J. D. McClure and R. E. Luna, 1981, *Report on a Workshop on Transportation Accident Scenarios Involving Spent Fuel*, SAND80-2012, TTC-0151, Transportation Analysis and Information Division, Sandia National Laboratories, Albuquerque, NM, February.

Wood, W. W. and W. H. Low, 1988, *Solute Geochemistry of the Snake River Plain Regional Aquifer System, Idaho and Eastern Oregon*, Professional Paper 1408-D, U.S. Geological Survey.

World Bank, 1994, *World Development Report, Infrastructure for Development*, Oxford University Press, New York, NY, June.

WSRC (Westinghouse Savannah River Company), 1995, *High-Level Waste System Plan Revision 5 (U)*, HLW-OVP-95-0031, Aiken, SC, April 26.

WSRC (Westinghouse Savannah River Company), 1994a, *Repository Emplacement Costs for Al-Clad HEU Spent Fuel (U)*, WSRC-TR-94-0230, Rev. 1, Aiken, SC, September 23.

WSRC (Westinghouse Savannah River Company), 1994b, *Life-Cycle Cost Analysis for Foreign Research Reactor Spent Nuclear Fuel Disposal (U)*, WSRC-TR-94-039, Aiken, SC, March 30.

WSRC (Westinghouse Savannah River Company), 1993a, *Technical Data Input, "Section 5.3.6, Geology,"* Draft 0, Rev. 0, Savannah River Site, Aiken, SC, October.

WSRC (Westinghouse Savannah River Company), 1993b, *Technical Data Input, "Section 5.3.8, Water Resources,"* Draft 0, Rev. 0, Savannah River Site, Aiken, SC, October.

Yocum, C., 1994, Concord Naval Weapons Station, CA, personal communication to R. L. Gotchy, Science Applications International Corporation, August 15 and 16.

Yohe, R., 1993, Idaho State Historical Preservation Office, personal communication to T. Rudolph, Science Applications International Corporation, Boise, ID.

Yuan, Y. C., S. Y. Chan, D. J. LePoire and R. Rothman, 1993, *RISKIND - A Computer Program for Calculating Radiological Consequences and Health Risks from Transportation of Spent Nuclear Fuel*, ANL/EAIS-06, Rev. 0, Argonne National Laboratory, Argonne, IL, February.

9. Glossary

Absorbed dose. The energy imparted by ionizing radiation per unit mass of irradiated material. The unit of absorbed dose is the rad.

Accident. An unplanned sequence of events that results in undesirable consequences.

Actinide. Any of a series of chemically similar, mostly synthetic, radioactive elements with atomic numbers ranging from actinium (89) through lawrencium (103).

Acute exposure. A single exposure to a toxic substance which may result in severe biological harm or death. Acute exposures are usually characterized as lasting no longer than a day.

Alpha-emitter. A radioactive substance that decays by releasing an alpha particle.

Alpha particle. A particle consisting of two protons and two neutrons, given off by the decay of many elements, including uranium, plutonium, and radon. Alpha particles cannot penetrate a sheet of paper. However, alpha emitting isotopes in the body can be very damaging.

As low as reasonably achievable (ALARA). The approach to radiation protection to manage and control exposures (both individual and collective) to the work force and to the general public to as low as is reasonable, taking into account social, technical, economic, practical, and public policy considerations. ALARA is not a dose limit but a process which has the objective of attaining doses as far below the applicable limits as is reasonably achievable.

Atomic Energy Act (AEA). A law passed in 1954 that placed nuclear production and control of nuclear materials within a civilian agency, originally the Atomic Energy Commission. The Atomic Energy Commission was replaced by the U.S. Nuclear Regulatory Commission, the U.S. Department of Energy, and predecessor agencies (i.e., ERDA, FERC).

Atomic number. The number of positively charged protons in the nucleus of an atom or the number of electrons on an electrically neutral atom.

Background radiation. Radiation from: (1) Naturally occurring radioactive materials which have not been technologically enhanced, (2) cosmic sources, (3) global fallout as it exists in the environment (such as from the testing of nuclear explosive devices), (4) radon and its progeny in concentrations or levels existing in buildings or the environment which have not been elevated as a result of current or prior activities, and (5) consumer products containing nominal amounts of radioactive material or producing nominal amounts of radiation.

Beta particle. A particle emitted in the radioactive decay of many radionuclides. A beta particle is identical with an electron. It has a short range in air and a low ability to penetrate other materials.

Canning. The process of placing spent nuclear fuel in canisters to retard corrosion, contain radioactive releases, or control geometry.

Cask. A heavily shielded massive container for holding nuclear materials during shipment.

Characterization. The determination of waste or spent nuclear fuel composition and properties, whether by review of process knowledge, nondestructive examination or assay, or sampling and analysis, generally done to determine appropriate storage, treatment, handling, transportation, and disposal requirements.

Chemical separation. A process for extracting uranium and plutonium from dissolved spent nuclear fuel and irradiated targets. The fission products that are left behind are high level wastes. Chemical separation is also known as reprocessing.

Cladding. The outer layer of metal over the fissile material of a nuclear fuel element. Cladding on the Department of Energy's spent fuel is usually aluminum, zirconium, or stainless steel.

Collective dose. The sum of the total effective dose equivalents of all individuals in a specified population. Collective dose is expressed in units of person-rem (or person-sievert).

Committed effective dose equivalent. The sum of the committed dose equivalents to various tissues in the body, each multiplied by the appropriate weighting factor. Committed effective dose equivalent is expressed in units of rem (or sievert), and will be accumulated during the fifty years following an intake of radioactive material into an individual's body.

Competitive fee. A fee that could be charged to foreign research reactor operators related to the estimated cost of spent nuclear fuel management and disposal outside the United States.

Conditioning. See stabilization (of spent nuclear fuel).

Contact-handled waste. Packaged waste whose external surface dose rate does not exceed 200 mrem per hour.

Contamination. The deposition of undesirable radioactive material on the surfaces of structures, areas, objects, or personnel.

Core. The central portion of a nuclear reactor containing the fuel elements, moderator, neutron poisons, and support structures.

Criticality. The conditions in which a system is capable of sustaining a nuclear chain reaction.

Cumulative impact. The impact on the environment which results from the incremental impact of the action when added to other past, present, and reasonably foreseeable future actions regardless of what agency or person undertakes such other actions. Cumulative impacts can result from individually minor but collectively significant actions taking place over a period of time.

Curie. The basic unit used to describe the intensity of radioactivity in a sample of material. The curie is equal to 37 billion disintegrations per second, which is approximately the rate of decay of 1 gram of the isotope radium-266. A curie is also a quantity of any radionuclide that decays at a rate of 37 billion disintegrations per second.

Decay (radioactive). Spontaneous disintegration of the nucleus of an unstable atom, resulting in the emission of particles and energy.

Decommissioning. Retirement of a nuclear facility, including decontamination and/or dismantlement.

Decontamination. Removal of unwanted radioactive or hazardous contamination by a chemical or mechanical process.

Degraded (spent nuclear fuel). See failed fuel.

Depleted uranium. Uranium that, through the process of enrichment, has been stripped of most of the uranium-235 it once contained, so that it has more uranium-238 than natural uranium. It is used as shielding, in some parts of nuclear weapons, and as a raw material for plutonium production.

Developed countries. Countries with high-income economies (World Bank, 1994).

Developing countries. Countries with other-than-high-income economies (World Bank, 1994).

Discounted dollars. Expressing income and expenditures that occur over time as if they occurred at a common point in time.

Disposal of fuel. Emplacement of fuel to ensure its isolation from the biosphere, with no intention of retrieval.

DOE Orders. Requirements internal to the U.S. Department of Energy (DOE) that establish DOE policy and procedures, including those for compliance with applicable laws.

Dose (or radiation dose). A generic term that means absorbed dose, dose equivalent, effective dose equivalent, committed effective dose equivalent, or total effective dose equivalent as defined elsewhere in this glossary.

Dose rate. The radiation dose delivered per unit time (e.g., rem per year).

Dry storage. Storage of spent nuclear fuel in environments where the fuel is not immersed in water for purposes of both cooling and shielding.

Ecology. The relationship of living things to one another and their environment, or the study of such relationships.

Effective dose equivalent. The summation of the products of the dose equivalent received by specified tissues of the body and the appropriate weighting factor. It includes the dose from radiation sources internal and/or external to the body. The effective dose equivalent is expressed in units of rem (or sievert).

Endangered species. Animals, birds, fish, plants, or other living organisms threatened with extinction by man-made or natural changes in their environment. Requirements for declaring a species endangered are contained in the Endangered Species Act.

Enriched uranium. Uranium that has greater amounts of the isotope uranium-235 than occurs naturally. Naturally occurring uranium is 0.72 percent uranium-235.

Environmental monitoring. The process of sampling and analysis of environmental media in and around a facility being monitored for the purpose of (1) confirming compliance with performance objectives and (2) early detection of any contamination entering the environment to facilitate timely remedial action.

Escalation. A real change in the price level of a particular good or service, unrelated to inflation.

Existing facilities. Facilities that existed at an active DOE site as of the Record of Decision for this Environmental Impact Statement.

Failed fuel. Spent nuclear fuel whose external cladding has cracked, pitted, corroded, or potentially allows the leakage of radioactive gases.

Fissile material. Any material fissionable by thermal (slow) neutrons; the two primary fissile isotopes are uranium-235 and plutonium-239.

Fission. The splitting or breaking of a nucleus into at least two other nuclei and the release of a relatively large amount of energy. Two or three neutrons are usually released during this type of transformation.

Fission products. The nuclei produced by fission of heavy elements, and their radioactive decay products.

Fissionable material. Commonly used as a synonym for fissile material, the meaning of this term has been extended to include material that can be fissioned by fast neutrons, such as uranium-238.

Fuel elements. Nuclear reactor fuel including both the fissile and the structural material serves as cladding.

Full-cost recovery fee. A fee that could be charged to foreign research reactor operators that recovers all costs incurred by the United States for management of their spent nuclear fuel.

Gamma ray. Very penetrating electromagnetic radiation of nuclear origin. Except for origin and energy level, identical to x-rays. Electromagnetic radiation frequently accompanying alpha and beta emissions as radioactive materials decay.

Geologic repository. A place to dispose of radioactive waste deep beneath the earth's surface.

Groundshine. The radiation dose received from radioactive material deposited on the ground's surface.

Half-life. The time in which one-half of the atoms of a particular radioactive substance disintegrate to another nuclear form.

Hazardous material. A substance or material in a quantity and form which may pose an unreasonable risk to health and safety or property when transported in commerce.

Hazardous substance. Any substance that when released to the environment in an uncontrolled or unpermitted fashion becomes subject to the reporting and possible response provisions of the Clean Water Act and the Comprehensive Environmental Response, Compensation, and Liability Act.

Hazardous waste. (1) Wastes that are identified or listed in 40 CFR 261.31 and 261.32. Source, special nuclear material, and by-product material as defined by the Atomic Energy Act of 1954, as amended, are specifically excluded from the term hazardous wastes. (2) As defined in RCRA, a solid waste, or combination of wastes, that because of its quantity, concentration, or physical, chemical, or infectious characteristics, may cause or significantly contribute to an increase in mortality or serious, irreversible, or incapacitating reversible illness or pose a substantial present or potential hazard to human health or the environment when improperly treated, stored, transported, or disposed of, or otherwise managed. (3) By-products of society that can pose a substantial or potential hazard to human health or the environment when improperly managed. Possesses at least one of four characteristics (ignitability, corrosivity, reactivity, or toxicity).

High-efficiency particulate air (HEPA) filter. A filter with an efficiency of at least 99.95 percent used to remove particles from air exhaust streams prior to releasing to the atmosphere.

Highly enriched uranium (HEU). Uranium with more than 20 percent of the uranium-235 isotope, used for making nuclear weapons and also as fuel for some isotope-production, research, naval propulsion, and power reactors.

High-level waste. The highly radioactive waste material that results from the reprocessing of spent nuclear fuel, including liquid waste produced directly from reprocessing and any solid waste derived from the liquid that contains a combination of transuranic and fission product nuclides in quantities that require permanent isolation. High-level waste may include the highly radioactive material that the NRC, consistent with existing law, determines by rule requires permanent isolation.

Inflation. A change in the nominal price level of all goods or services, unrelated to the real escalation of a particular good or service.

Isotopes. Different forms of the same chemical element that differ only by the number of neutrons in their nucleus. Most elements have more than one naturally occurring isotope. Many more isotopes have been produced in reactors and scientific laboratories.

Latent cancer fatalities (LCF). Deaths occurring at later years from radiation-induced cancers.

Levelization. Conversion of a stream of values that vary at a uniform rate over time to a constant value over the same period of time.

Life cycle costs. All costs except the cost of personnel occupying the facility incurred from the time that space requirement is defined until the facility passes out of the government's hands.

Low enriched uranium (LEU). Uranium enriched until it consists of up to 20 percent uranium-235. Used as nuclear reactor fuel.

Low-level waste. A catchall term for any radioactive waste that is not spent fuel, high-level, or transuranic waste.

Management (spent nuclear fuel). Emplacing, operating, and administering facilities, transportation systems, and procedures in order to ensure safe and environmentally responsible handling and storage of spent nuclear fuel pending (and in anticipation of a decision on ultimate disposition. Spent nuclear fuel management also includes activities such as stabilization, examination/characterization, processing or chemical separation, and research and development; including activities that may be necessary to prepare spent nuclear fuel for ultimate disposition.

Maximally exposed individual (MEI). A theoretical individual living at the site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point.

Maximally exposed worker. A marine transport worker, port worker, ground transport worker, or onsite radiation worker who could receive the maximum radiation exposure in a given situation.

Maximum contaminant level (MCL). The maximum permissible levels of a contaminant in water which is delivered to the free flowing outlet of the ultimate user of a public water system, except in the case of turbidity where the maximum permissible level is measured at the point of entry to the distribution system. Contaminants added to the water under the circumstances controlled by the user, except those resulting from corrosion of piping and plumbing caused by water quality, are excluded from this definition.

Metric tons of heavy metal (MTHM). Quantities of unirradiated and spent nuclear fuel and targets are traditionally expressed in terms of metric tons of heavy metal (typically uranium), without the inclusion of other materials, such as cladding, alloy materials, and structural materials. A metric ton is 1,000 kilograms, which is equal to about 2,200 pounds.

National Environmental Policy Act. A Federal law, enacted in 1970, that requires the Federal government to consider the environmental impacts of, and alternatives to, major proposed actions in its decisionmaking processes. Commonly referred to by its acronym, NEPA.

Natural phenomena accidents. Accidents that are initiated by phenomena such as earthquakes, tornadoes, floods, etc.

Nearest public access individual (NPAI). A theoretical individual located at the point of nearest public access to a DOE facility, usually during an accident situation.

Net present value. The value of a series of future income and expense streams brought forward to the present at the discount rate.

Neutron. Uncharged elementary particles with a mass slightly greater than that of the proton, and found in the nucleus of every atom heavier than hydrogen.

Nonproliferation. Efforts to prevent or slow the spread of nuclear weapons and the materials and technologies used to produce them.

Normal operation. All normal conditions and those abnormal conditions that frequency estimation techniques indicate occur with a frequency greater than 0.1 events per year.

Nuclear fuel. Materials that are fissionable and can be used in nuclear reactors.

Plutonium. A manmade fissile element. Pure plutonium is a silvery metal that is heavier than lead. Material rich in the plutonium-239 isotope is preferred for manufacturing nuclear weapons, although any plutonium can be used. Plutonium-239 has a half-life of 24,000 years.

Population dose. See collective dose.

Probable maximum flood. The largest flood for which there is any reasonable expectancy in a specific area. The probable maximum flood is normally several times larger than the largest flood of record.

Processing (of spent nuclear fuel). Applying a chemical or physical process designed to alter the characteristics of the spent nuclear fuel matrix.

Public. Anyone outside the DOE site boundary at the time of an accident or during normal operation.

PUREX. An acronym for Plutonium-Uranium Extraction, the name of the chemical process usually used to reprocess spent nuclear fuel and irradiated targets.

Rad. The special unit of absorbed dose. One rad (0.01 gray) is equal to an absorbed dose of 100 ergs/gram.

Radiation (ionizing). Energy transferred through space or other media in the form of particles or waves. In this document, we refer to ionizing radiation which is capable of breaking up atoms or molecules. The splitting, or decay, of unstable atoms emits ionizing radiation.

Radioactive waste. Waste that is managed for its radioactive content; solid, liquid or gaseous material that contains radionuclides regulated under the AEA of 1954, as amended and of negligible economic value considering costs of recovery.

Radioactivity. The spontaneous emission of radiation from the nucleus of an atom. Radionuclides lose particles and energy through this process of radioactive decay.

Region of influence. Region in which the principal direct and indirect socioeconomic effects of actions are likely to occur and are expected to be of consequence for local jurisdictions.

Regulated substances. A general term used to refer to materials other than radionuclides that may be regulated by other applicable Federal, State, (or possibly local) requirements.

rem. Roentgen Equivalent Man which is a unit of dose equivalent. Dose equivalent in rem is numerically equal to the absorbed dose in rad multiplied by a quality factor, distribution factor and any other necessary modifying factor (1 rem = 0.01 sievert).

Reprocessing (spent nuclear fuel). See chemical separation.

Risk. Quantitative expression of possible loss that considers both the probability that a hazard causes harm and the consequences of that event.

Saltstone. Low-radioactivity fraction of high-level waste formed into a concrete block at the Savannah River Site.

Source material. (1) Uranium, thorium, or any other material that is determined by the Nuclear Regulatory Commission pursuant to the provisions of the Atomic Energy Act of 1954, Section 61, to be source material; or (2) ores containing one or more of the foregoing materials, in such concentration as the Nuclear Regulatory Commission may by regulation determine from time-to-time [Atomic Energy Act 11(z)].

Special nuclear material. (1) Plutonium, uranium enriched in the isotope 233 or in the isotope 235, and any other material that the Nuclear Regulatory Commission, pursuant to the provisions of the Atomic Energy Act of 1954, Section 51, determines to be special nuclear material, but does not include source material; or (2) any material artificially enriched by any of the foregoing, but does not include source material.

Spent nuclear fuel. Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated.

Stabilization (of spent nuclear fuel). Actions taken to further confine or reduce the hazards associated with spent nuclear fuel, as necessary for safe management and environmentally responsible storage for extended periods of time. Activities which may be necessary to stabilize spent nuclear fuel include canning, processing, and passivation.

Storage. The collection and containment of waste or spent nuclear fuel in such a manner as not to constitute disposal of the waste or spent nuclear fuel for the purposes of awaiting treatment or disposal capacity (i.e., not short-term accumulation).

Surface water. All waters that are open to the atmosphere and subject to surface runoff. All waters naturally open to the atmosphere (rivers, lakes, reservoirs, streams, impoundments, seas, estuaries, etc.) and all springs, wells, or other collectors that are directly influenced by surface water.

Target. A tube, rod, or other form containing material that, on being irradiated in a nuclear reactor would produce a designed end product (i.e., uranium-238 produces plutonium-239 and neptunium-237 produces plutonium-238).

Target material. Residual material that is left after a target has been irradiated and dissolved, and the end product has been removed. In this EIS, target material contains enriched uranium and fission products.

Total effective dose equivalent. The sum of the deep dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures).

Type B packaging. Packaging for radioactive material which meets the standards for Type A packaging and, in addition, meets the standards for the hypothetical accident conditions of transport as prescribed in 49 Code of Federal Regulations Part 173.398(c). This includes spent fuel casks.

Ultimate disposition. The final step in which a material is either processed for some use or disposed of.

Undiscounted dollars. Expressing income and expenditures in the year they occur, not at some common point in time.

Uranium. The basic material for nuclear technology. It is a slightly radioactive naturally occurring heavy metal that is more dense than lead. Uranium is 40 times more common than silver.

Vitrification. The process of immobilizing waste that produces a glass-like solid that permanently captures the radioactive materials.

Vulnerabilities. Conditions or weaknesses that may lead to radiation exposure to the public, unnecessary or increased exposure to the workers, or release of radioactive materials to the environment.

Waste classification. Wastes are classified according to 10 CFR § 61.55 for the purpose of near surface disposal to three classes: A, B, and C. Class C waste represents the waste that must meet the most rigorous requirements on waste form to ensure stability and additional measures at the disposal facility to protect against inadvertent intrusion.

Waste management. The planning, coordination, and direction of those functions related to generation, handling, treatment, storage, transportation, and disposal of waste, as well as associated surveillance and maintenance activities.

Waste minimization. An action that economically avoids or reduces the generation of waste by source reduction or reduces the toxicity of hazardous waste, improving energy usage, or by recycling. This action will be consistent with the general goal of minimizing present and future threats to human health, safety, and the environment.

Wet storage. Storage of spent nuclear fuel in a pool of water, generally for the purposes of both cooling and worker shielding.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix A
Environmental Justice Analysis

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

List of Tables

Appendix A
Environmental Justice Analysis

A.1 Introduction

Executive Order 12898, *Federal Actions to Address Environmental Justice in Minority Populations and Low Income Populations,* directs Federal agencies to identify and address, as appropriate, disproportionately high and adverse human health or environmental effects of their programs, policies, and activities on minority and low-income populations. Executive Order 12898 also directs the Administrator of the Environmental Protection Agency to convene an interagency Federal Working Group on Environmental Justice. The Working Group is directed to provide guidance to Federal agencies on criteria for identifying disproportionately high and adverse human health or environmental effects on minority and low-income populations. The Working Group has not yet issued the guidance directed by Executive Order 12898, although it has developed working draft definitions. The definitions used in this analysis are based on the draft working definitions. Further, in coordination with the Working Group, DOE is in the process of developing internal guidance on implementing the Executive Order. Because both the Working Group and DOE are still in the process of developing guidance, the approach taken in this analysis may depart somewhat from whatever guidance is eventually issued.

This appendix addresses environmental justice for the acceptance of foreign research reactor spent nuclear fuel containing uranium enriched in the United States. Analyses of environmental justice concerns are provided in three areas: (1) potential ports of entry, (2) potential transportation routes from candidate ports of entry to interim management sites, and (3) areas surrounding potential interim management sites. These analyses lead to the conclusion that the alternatives analyzed in this Environmental Impact Statement (EIS) would result in no disproportionately high and adverse effects on minority populations or low-income communities surrounding the candidate ports, transport routes, or interim management sites.

A.2 Concerns and Definitions

Public comments show a widespread concern for public health and safety because spent nuclear fuel is radioactive. Two related environmental documents (DOE, 1995 and DOE, 1994) have been published recently which address the safety and potential health issues due to transportation and storage of spent nuclear fuel. Analyses of radiological health effects in those documents as well as this EIS (see Chapter 4) demonstrate that the expected health effects are small. In the case of spent nuclear fuel from foreign research reactors, no fatalities are expected due to radiological exposure or traffic accidents. No significant health effects are expected for the general population. Consequently, there would be no disproportionately high or adverse human health effects imposed on any population segment. In the sections below, minority and low-income populations are identified in the areas near potential candidate ports of entry, potential interim management sites, and potential transportation routes. The 1990 census data were used in this appendix as the basis of the analysis (DOC, 1992). This allows equal comparison of data between ports, sites, and routes in different states.

The analysis uses the following draft definitions:

- *Minority* — Individuals classified by the U.S. Bureau of the Census as Negro/Black/African American, Hispanic, Asian and Pacific Islander, American Indian, Eskimo, Aleut, and other non-White persons. The minority population in an affected area is the number of individuals residing in the area who are members of a minority group.
- *Low-Income Community* — An area for which the median household income is 80 percent or below the median household income for the metropolitan statistical area (urban) or county (rural). While "80 percent" is used in this analysis based on definitions used by the U.S. Department of Housing and Urban Development, this percentage may change in the final guidelines under preparation by the Working Group and the Department of Energy.
- *Disproportionately High and Adverse Human Health Effects* — Any human health effects, including cumulative or synergistic effects, on minority or low-income populations which substantially exceed generally accepted levels of risk. This is a draft definition prepared by the Working Group which might change during preparation of the final guidelines.
- *Substantially Affect Human Health* — To impact human health such that there is a measurable incidence of any specific illness, disease, or disorder significantly higher than the national average. This is also a draft definition developed by the Working Group which might change during preparation of final guidelines.

A.3 Environmental Justice in Areas Near the Candidate Ports of Entry

Under normal port activities associated with receipt of the spent nuclear fuel shipments — including harbor activities, unloading the ship, transfer of the spent nuclear fuel casks to truck or train, and movement out of the port city — the dominant radiological impacts were shown in Section 4.2.2 to be the exposures received by the workers in the immediate vicinity of the shipping cask. These individuals include inspectors, shipping cask handlers, and truck drivers. Since the intensity of the radiation from the cask falls off with distance, the doses that might be received by other workers and members of the general population can theoretically be calculated, but would not generally be measurable or distinguishable from natural background radiation.

Potential radiological impacts to people residing near the port are associated with low probability (less than one in a million) accidents that are so severe that the spent nuclear fuel casks rupture and a fire would burn long enough around the cask that some of the radioactive material would be released. In this case, some of the radioactive spent nuclear fuel might be vaporized and lifted by the heat of the fire and carried downwind of the accident location. Where and how far this radioactive material would go before being deposited on the ground would depend on how high the heat from the fire lofts it and the particular weather conditions at the time. Most of this vaporized spent nuclear fuel would be expected to be deposited in the first few miles downwind of the fire but small amounts could be carried out for several tens of miles.

Because the particular details of both the accident conditions (such as the severity of the fire) and the weather conditions at the time of an accident could vary widely, a range of accident conditions and wind directions, wind speeds, and other weather conditions were examined during the evaluation of accident effects (see Section 4.2.2.3). Population impact evaluations were performed for distances out to 80 km (50 mi). Risks of latent cancer deaths were found to range from about 0.003 to 0.000003 latent cancer fatalities (LCF). No latent cancer fatalities would be expected due to accidents at ports.

Containerized spent nuclear fuel casks shipped under the proposed policy would be transferred from the ship at commercial or military ports by personnel experienced in handling containerized cargo, and shipped by truck or rail to one of the five candidate interim management sites. Candidate ports may handle thousands of standard containers each month, unloaded from vessels which can carry up to several thousand casks. The number of casks to be handled would be small in comparison to routine cargo handling, thus having a negligible impact on normal port activities.

As part of the environmental justice analysis, distributions of minority populations and low-income households surrounding candidate ports of entry were estimated from 1990 census data. Although radiological health effects resulting from an accident are calculated at distances up to 80 km (50 mi), the largest radiological effects would usually be expected to occur within roughly a 16-km (10-mi) radius of the accident site. Thus, the distribution of minority and low-income populations is described for circular areas defined by a 16-km (10-mi) radius, centered at each candidate port of entry.

A.3.1 Distribution of Minority Populations Near the Candidate Ports

The minority population characteristics within 16 km (10 mi) of candidate ports of entry for foreign research reactor spent nuclear fuel are presented in Table A-1. For comparison, this table lists minority population features for regions surrounding the ports and for counties which lie partially within the 16-km (10-mi) radius centered at the port. Population characteristics shown in the table were extracted from 1990 census data available from the U.S. Bureau of the Census. The data resolves population characteristics at the "block group level," which generally consists of between 250 and 550 housing units.

With the exception of the Port of Wilmington and 2 military ports, MOTSU (Military Ocean Terminal, SUnny Point) and NWS (Naval Weapons Station) Concord, the percentage of minority populations residing within 16 km (10 mi) of candidate ports exceeds the percentage of minority populations residing within the state. Similarly, the percentage of minority populations residing near the candidate ports exceeds the percentage of minorities residing in counties surrounding the candidate ports. Ports at MOTSU, NWS Concord, Portsmouth, and Newport News are exceptions with larger percentages of minority populations in the surrounding counties.

The racial and ethnic composition of minority populations residing near the candidate ports is shown in Table A-2. In the case of candidate ports located on the east coast, African Americans compose the largest portion of the minority population. Minority populations residing near the candidate ports on the west coast are comprised of a more uniform mixture of African Americans, Asians, Hispanics, and Native Americans. The minority population residing near the Port of Galveston on the Gulf of Mexico is predominately African American and Hispanic.

The spatial distribution of minority populations residing within 16 km (10 mi) of each of the candidate ports is shown in the maps of those ports as presented in Figures A-1 to A-11. The circle shown in each figure has a 16-km (10-mi) radius, centered on the port. As indicated in the legend of each figure, geographical areas are shaded according to the percentage of minority population within the area. Resolution in the figures is at the census block group level. Due to variations in the populations of block groups, the geographical size of any particular block group area is not necessarily proportional to the numerical population. As an example, for ease of enumeration, the U.S. Bureau of the Census may define block group boundaries which actually extend into oceans, bays, or lakes. This allows inclusion in the census data of individuals who reside on boats or offshore houses, a situation particularly predominant in locations such as Galveston (see Figure A-3).

Table A-1 Minority Populations Residing Near the Candidate Ports

Candidate Port	Total Population Residing within 16 km of Port	Minority Population Residing within 16 km of Port	% Minority Population Residing within 16 km of Port	Total Population Residing in Surrounding Counties	Minority Population Residing in Surrounding Counties	% Minority Population Residing in Surrounding Counties	Population Residing in Surrounding State(s)	Minority Population Residing in Surrounding State(s)	% Minority Population Residing in Surrounding State(s)
Charleston, SC:									
Wando Terminal	233,424	82,271	35.2	423,815	145,534	34.3	3,486,703	1,094,792	31.4
NWS Terminal	209,188	73,437	35.1	423,815	145,534	34.3	3,486,703	1,094,792	31.4
Galveston, TX	73,322	36,375	49.6	217,445	72,133	33.2	19,986,510	6,665,631	33.4
Hampton Roads, VA:									
Newport News	430,757	161,317	37.4	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Norfolk	681,864	300,179	44.0	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Portsmouth	665,700	248,099	37.3	1,010,296	400,061	39.6	6,187,358	1,484,501	24.0
Jacksonville, FL	334,212	123,336	36.9	758,647	203,833	26.9	12,937,926	3,449,230	26.7
MOTSU, NC	7,995	1,496	18.7	50,985	9,835	19.3	6,628,637	1,651,356	24.9
NWS Concord, CA	381,070	110,969	29.1	1,145,248	375,442	32.8	29,760,021	12,666,060	42.6
Portland, OR	356,064	54,704	15.4	1,395,233	138,500	9.9	2,842,321	261,730	9.2
Savannah, GA	155,166	80,361	51.8	344,677	128,206	37.2	9,964,919	3,023,249	30.3
Tacoma, WA	511,575	85,341	16.7	2,123,421	347,788	16.4	4,866,692	637,561	13.1
Wilmington, NC	115,057	27,301	23.7	200,124	44,757	22.4	6,628,637	1,651,356	24.9

Table A-2 Racial and Ethnic Composition for Populations Residing Within 16 km of the Candidate Ports

Candidate Port	Total Pop.	Total Minority Pop.	% Minority Pop.	Amer. Indian, Eskimo or Aleut Pop.	% Amer. Indian, Eskimo or Aleut	Asian or Pacific Islander Pop.	% Asian or Pacific Islander	African Amer. Pop.	% African Amer.	Hispanic Origin Pop.	% Hispanic Origin	Other Race	% Other Race	White	% White
Charleston, SC:															
Wando Terminal	233,424	82,271	35.2	531	0.2	1,804	0.8	76,783	32.9	3,042	1.3	109	0.05	151,143	64.8
NWS Terminal	209,188	73,437	35.1	767	0.4	3,496	1.7	64,961	31.1	4,099	2.0	115	0.05	135,751	64.9
Galveston, TX	73,322	36,375	49.6	262	0.4	1,271	1.7	19,737	26.9	15,012	20.5	90	0.12	36,946	50.4
Hampton Roads, VA:															
Newport News	430,757	161,317	37.4	1,932	0.4	7,872	1.8	138,920	32.3	12,300	2.9	292	0.07	269,441	62.6
Norfolk	681,864	300,179	44.0	2,971	0.4	10,697	1.6	270,729	39.7	15,308	2.2	471	0.07	346,410	50.8
Portsmouth	665,700	248,099	37.3	2,763	0.4	9,612	1.4	221,200	33.2	14,069	2.1	453	0.07	322,815	48.5
Jacksonville, FL	334,212	123,336	36.9	960	0.3	5,456	1.6	108,641	32.5	8,149	2.4	128	0.04	210,815	63.1
MOTSU, NC	7,995	1,496	18.7	32	0.4	7	0.1	1,359	17.0	90	1.1	6	0.08	6,498	81.3
NWS Concord, CA	381,070	110,969	29.1	2,769	0.7	42,788	11.2	26,452	6.9	38,498	10.1	460	0.12	270,102	70.9
Portland, OR	356,064	54,704	15.4	4,086	1.1	12,617	3.5	27,012	7.6	10,632	3.0	355	0.10	301,359	84.6
Savannah GA	155,166	80,361	51.8	370	0.2	1,578	1.0	76,583	49.4	1,734	1.1	94	0.06	74,805	48.2
Tacoma, WA	511,575	85,341	16.7	7,095	1.4	28,321	5.5	32,687	6.4	16,779	3.3	457	0.09	426,231	83.3
Wilmington, NC	115,057	27,301	23.7	534	0.5	587	0.5	25,360	22.0	796	0.7	22	0.02	87,755	76.3

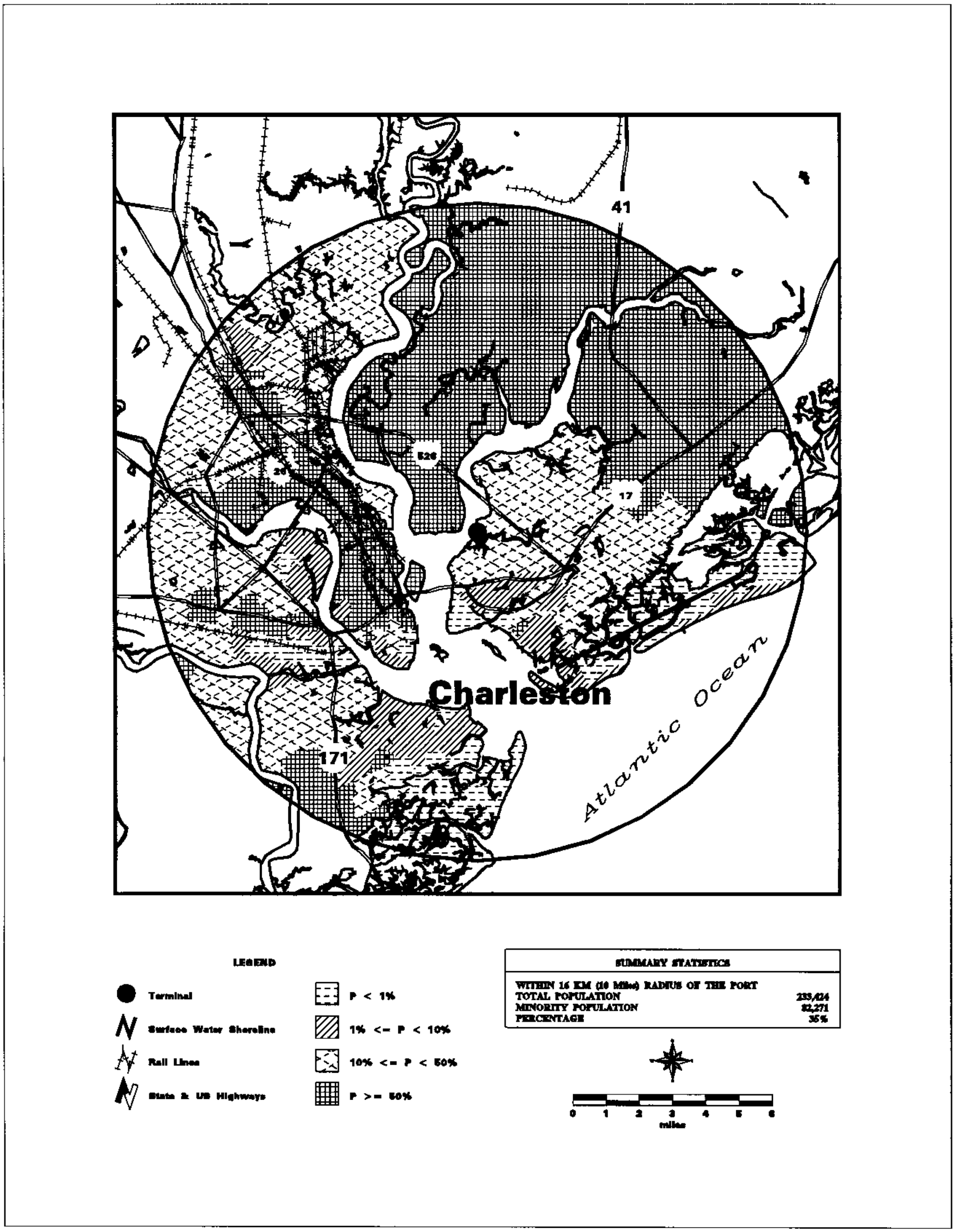


Figure A-1 Distribution of the Minority Population Residing within 16 km of the Wando Terminal, Port of Charleston, South Carolina

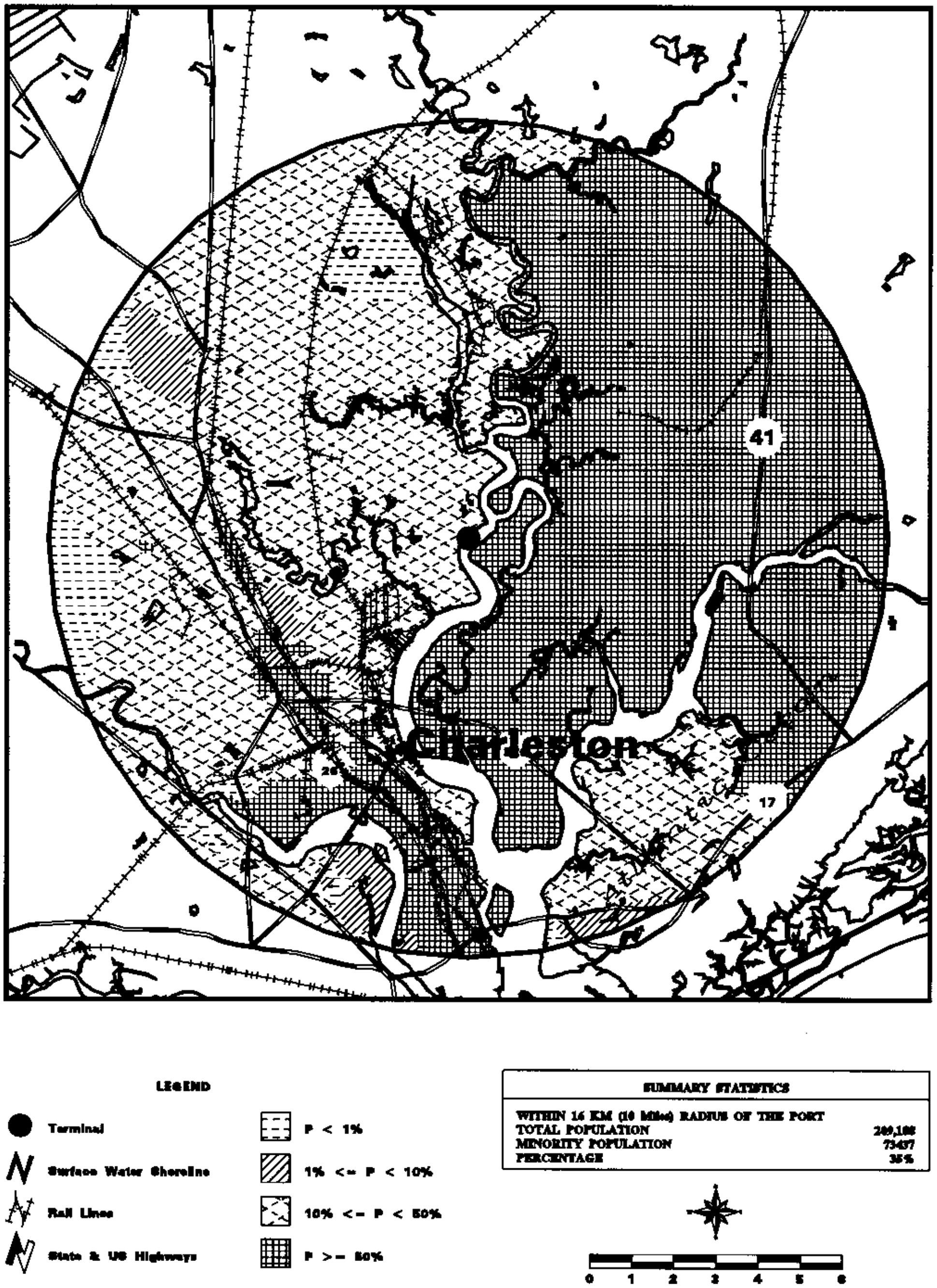


Figure A-2 Distribution of the Minority Population Residing within 16 km of the NWS Charleston, South Carolina

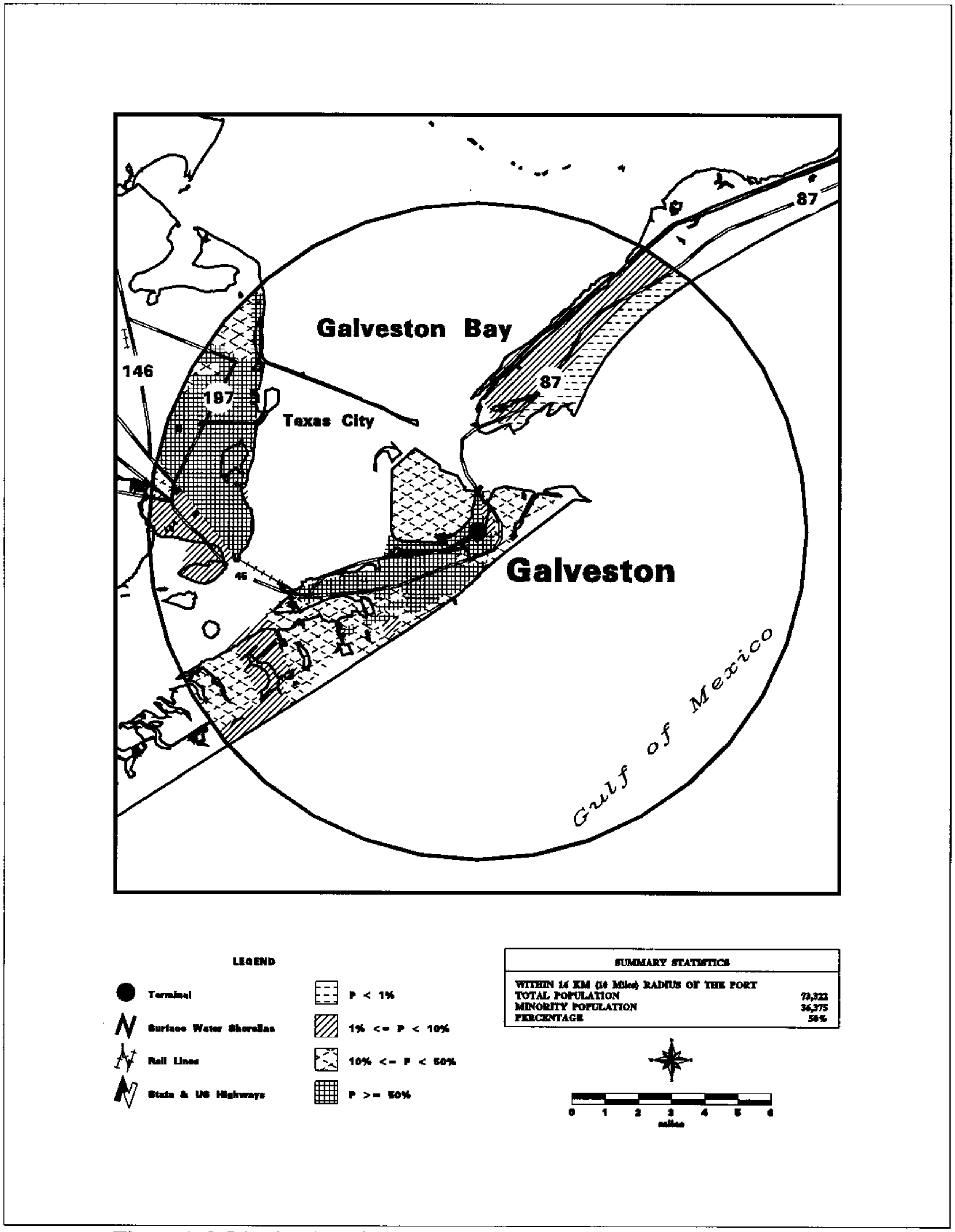


Figure A-3 Distribution of the Minority Population Residing within 16 km of the Port of Galveston, Texas

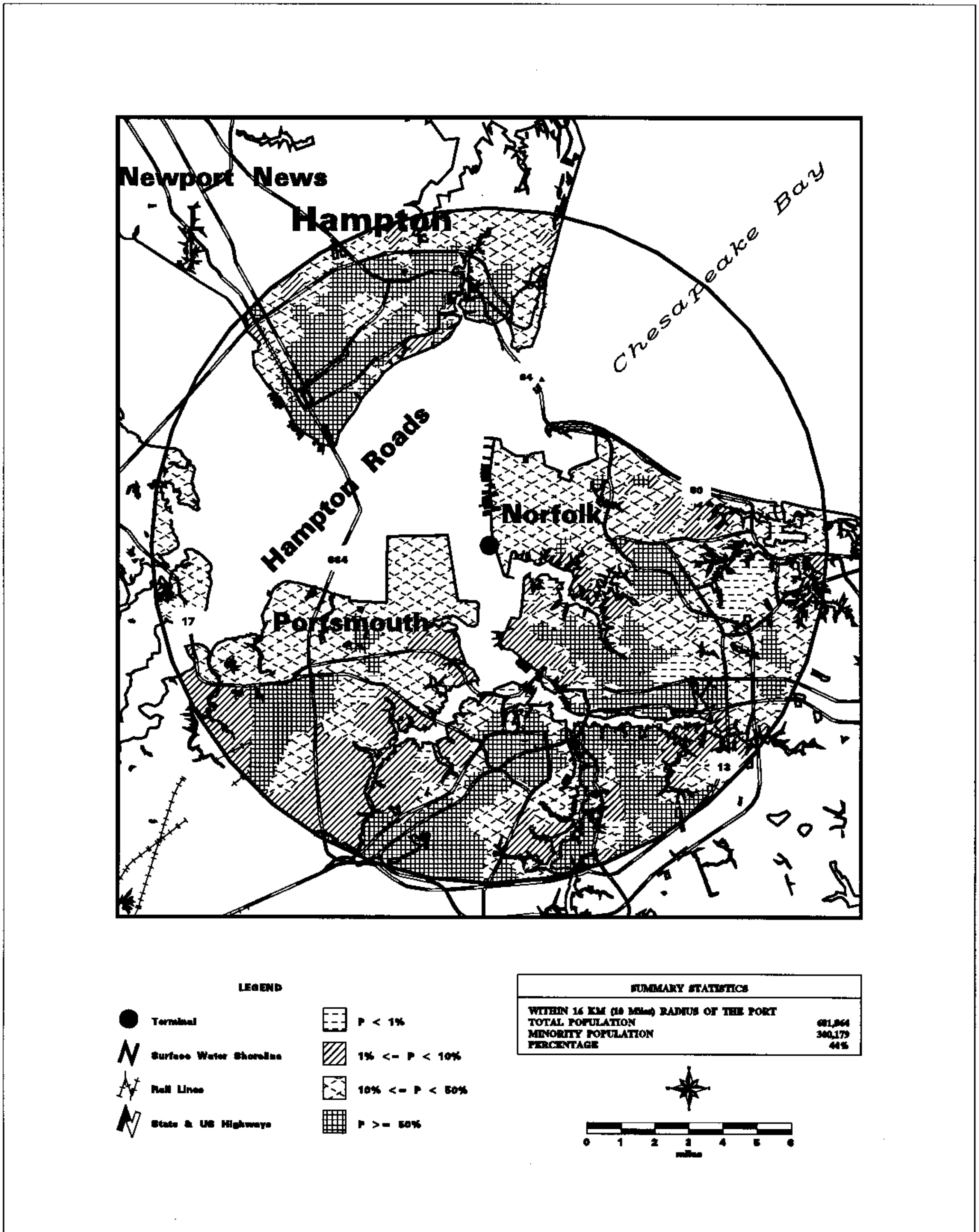


Figure A-4 Distribution of the Minority Population Residing within 16 km of the Port of Hampton Roads: Newport News, Norfolk, and Portsmouth, Virginia Terminals

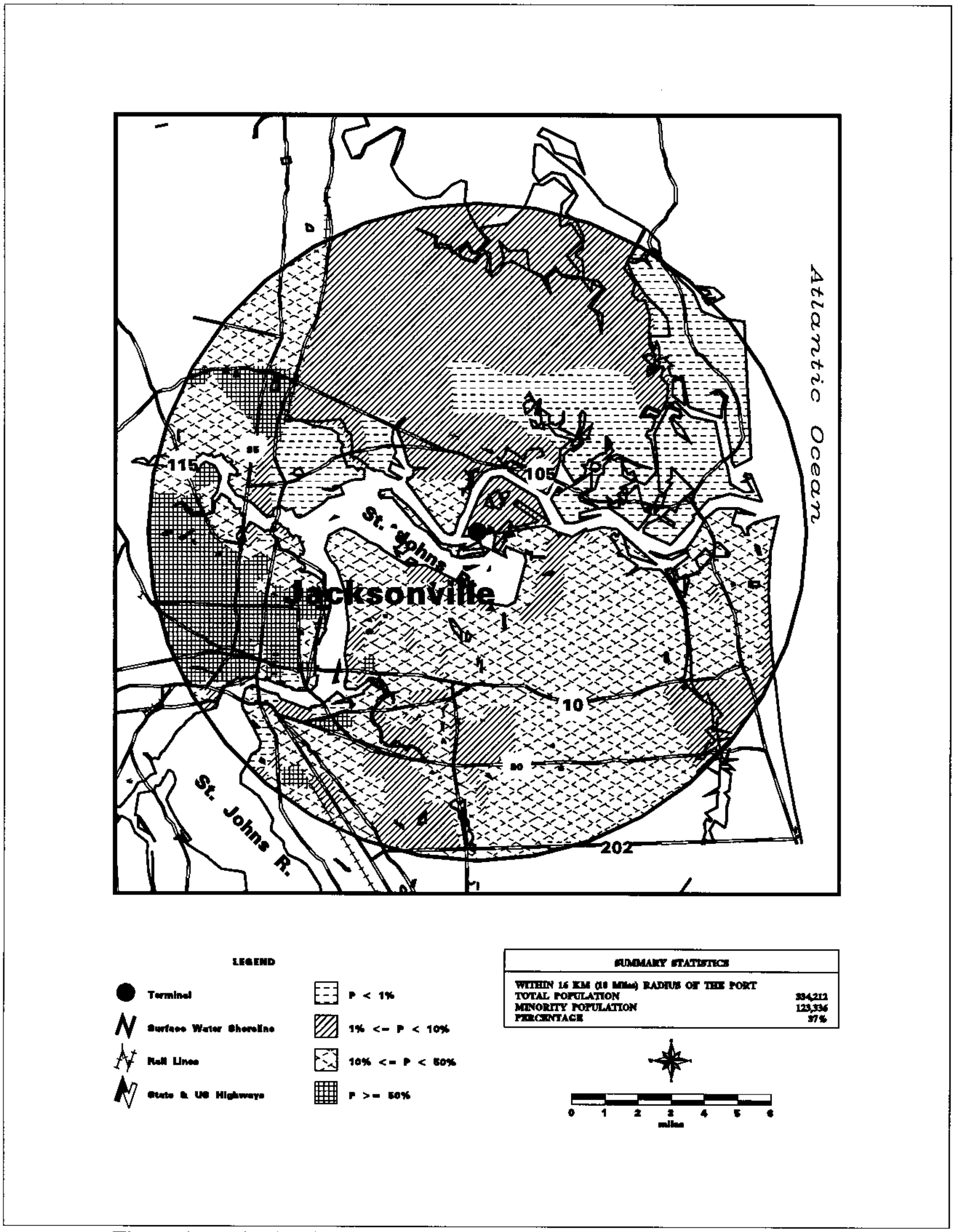


Figure A-5 Distribution of the Minority Population Residing within 16 km of the Port of Jacksonville, Florida

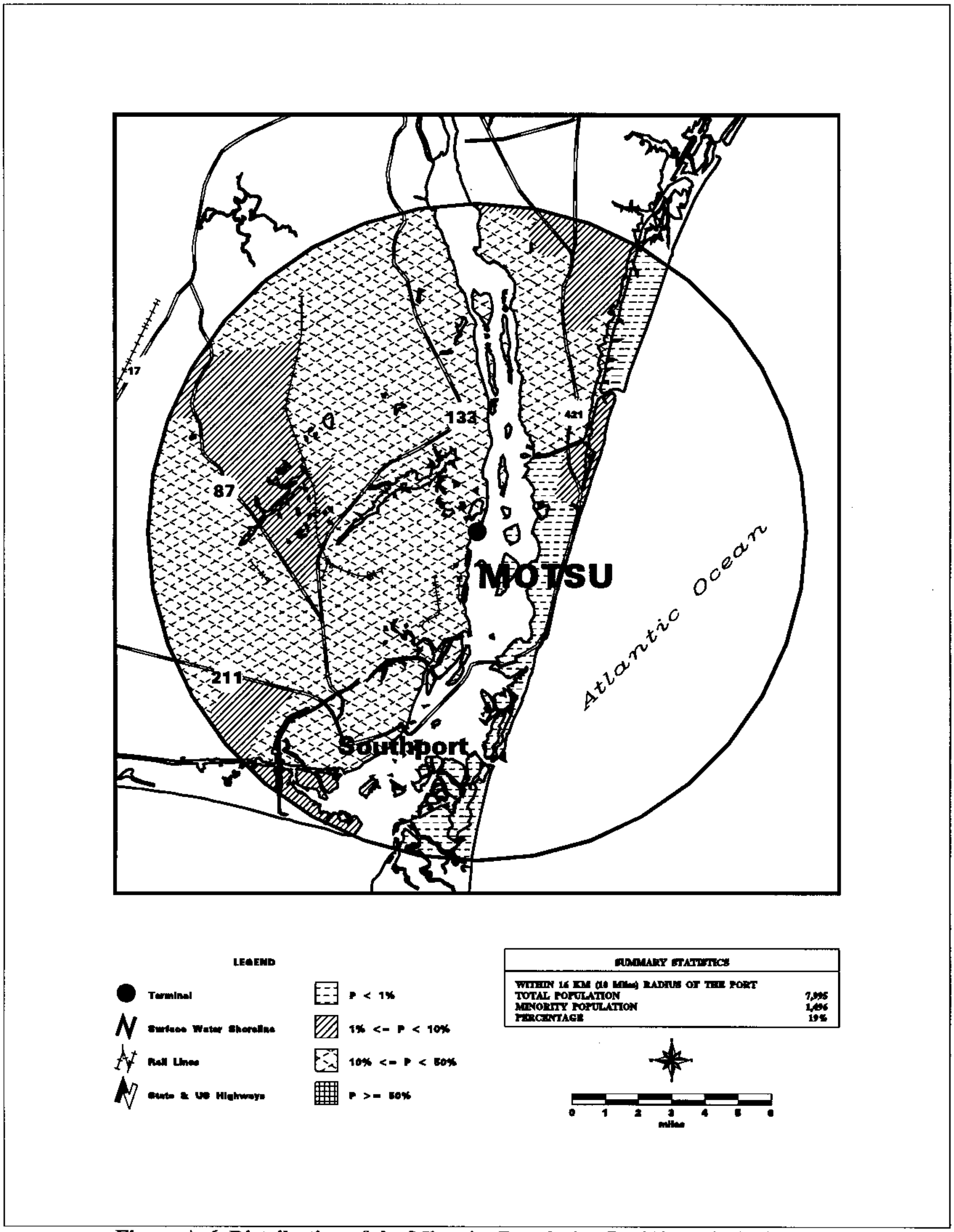


Figure A-6 Distribution of the Minority Population Residing within 16 km of the Port of MOTSU, North Carolina

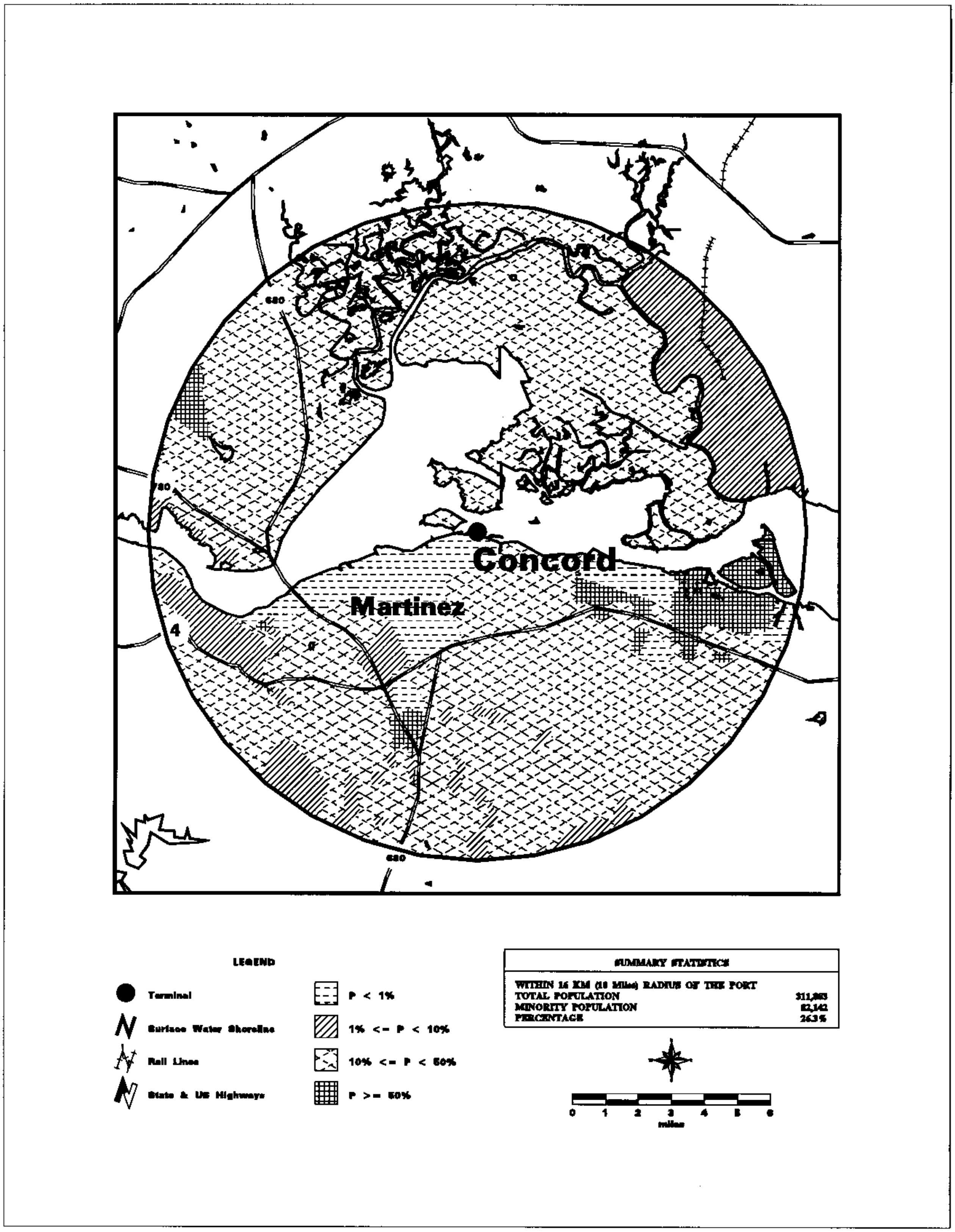


Figure A-7 Distribution of the Minority Population Residing within 16 km of the Port of NWS Concord, California

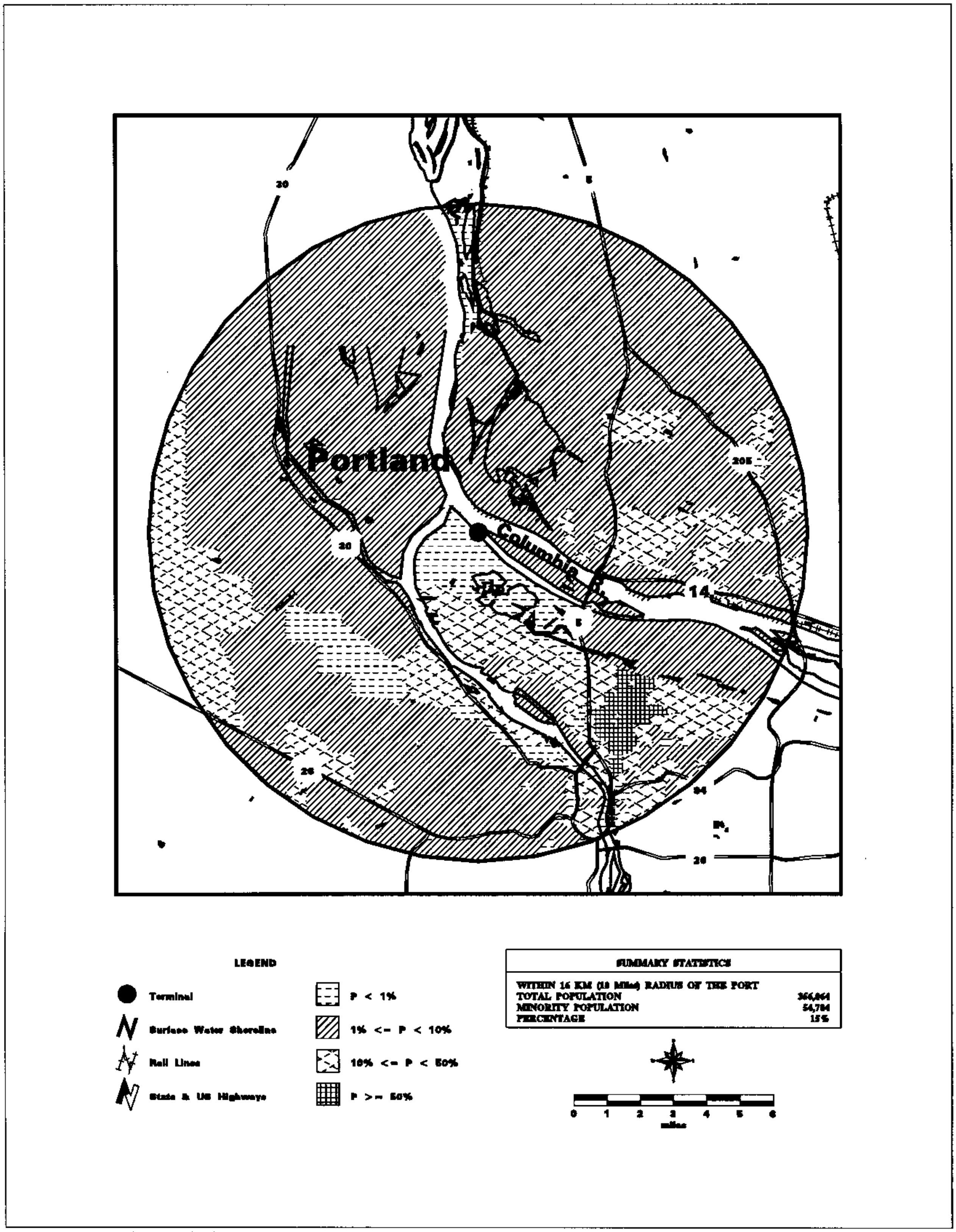


Figure A-8 Distribution of the Minority Population Residing within 16 km of the Port of Portland, Oregon

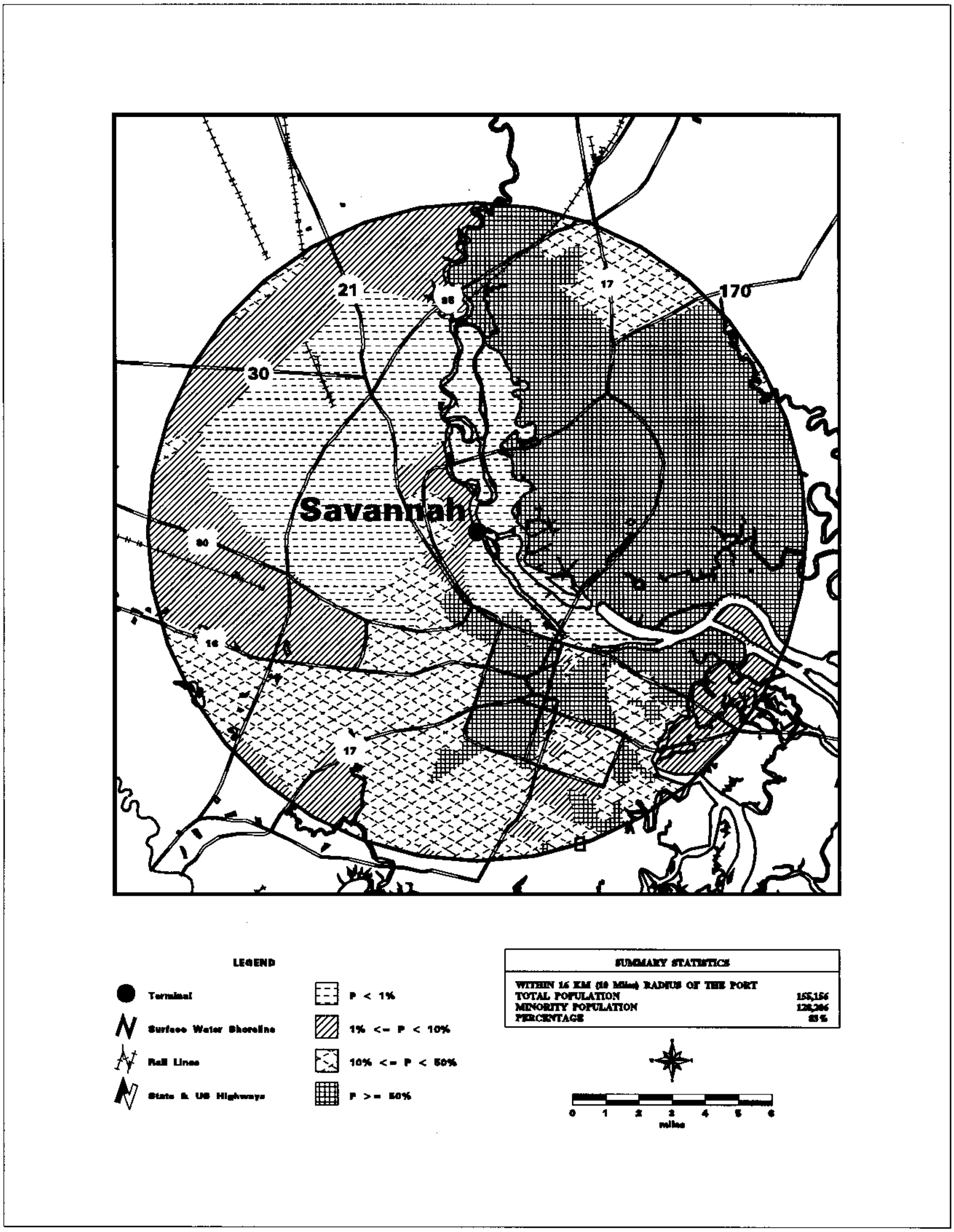


Figure A-9 Distribution of the Minority Population Residing within 16 km of the Port of Savannah, Georgia

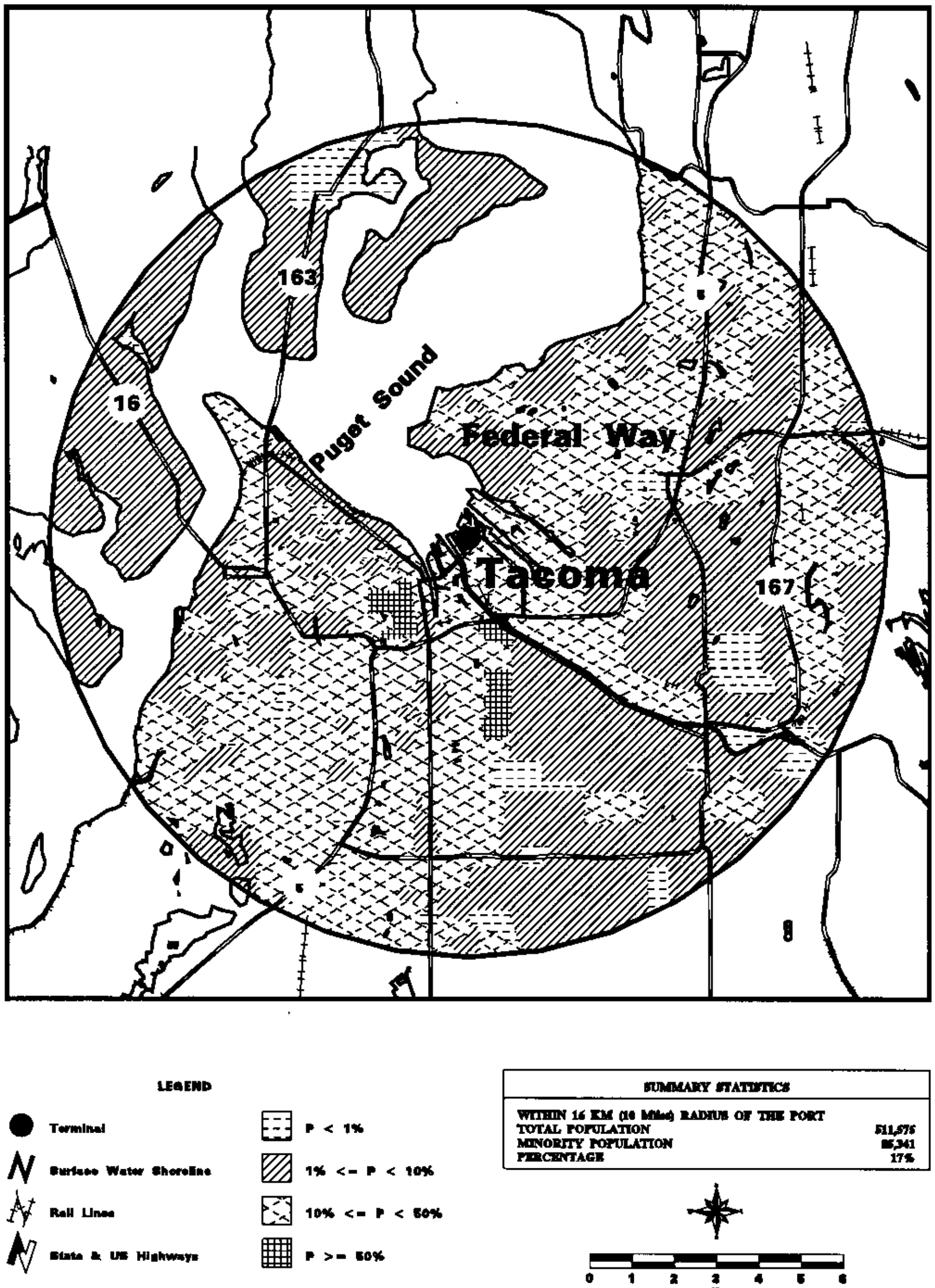


Figure A-10 Distribution of the Minority Population Residing within 16 km of the Port of Tacoma, Washington

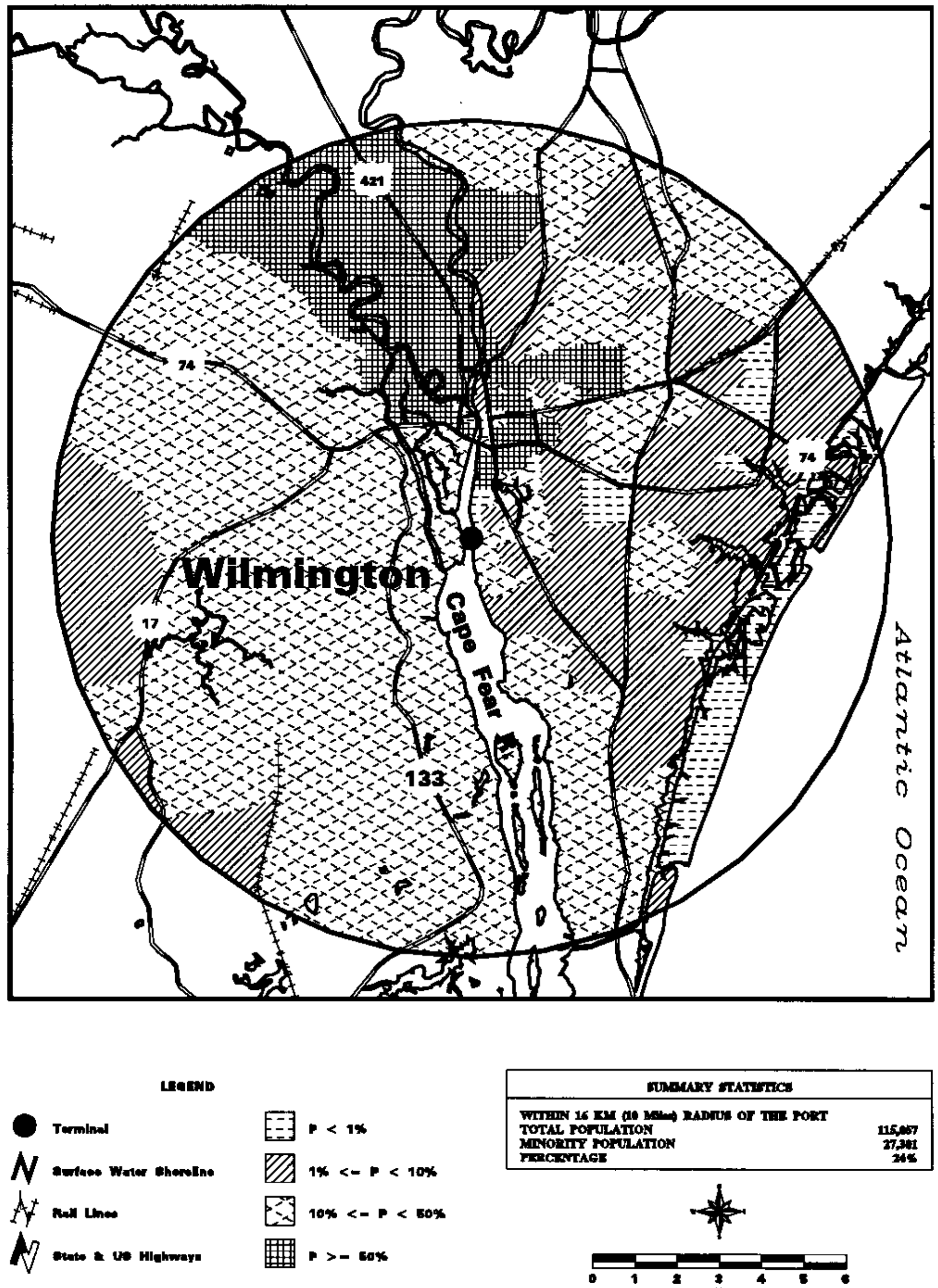


Figure A-11 Distribution of the Minority Population Residing within 16 km of the Port of Wilmington, North Carolina

A.3.2 Distribution of Low-Income Households Near the Candidate Ports

The number of low-income households near the candidate ports is shown in Table A-3. Except for the ports of MOTSU and Hampton Roads, the percentage of low-income households immediately surrounding the port is larger than the percentage of low-income households in the surrounding counties. Similarly, for most of the candidate ports, the percentage of low-income households near the port exceeds the percentage of low-income households in the surrounding state, although the ports of Charleston, MOTSU, Newport News, and NWS Concord are exceptions.

Distributions of low-income households near the candidate ports are shown in the maps of the ports presented in Figures A-12 through A-22. In these figures, geographical areas defined by census block group boundaries are shaded according to the percentage of low-income households within the block group. Since the number of households within a block group varies, the size of a shaded area is not necessarily proportional to the population within that area.

A.4 Environmental Justice Along Transportation Routes

The dominant radiological impacts associated with the normal or incident-free (accident-free) transportation activities would be the exposures received by the workers in the immediate vicinity of the cask, principally the truck drivers or train personnel. These individuals would be the only people receiving a measurable exposure during a routine spent nuclear fuel shipment.

The dose received by an individual near a spent nuclear fuel cask during shipment would be proportional to both the distance from the cask and the time of exposure. As discussed in Chapter 4 and Appendix E, the radiation dose rate from a cask containing spent nuclear fuel decreases with distance from the cask. Individuals living along the transportation routes would therefore be expected to receive low exposures because of both their distance from the cask and their short time of their exposure. While it is possible to make estimates of the collective dose of the population along a route, as in Chapter 4 and Appendix E, these minuscule doses would only be meaningful in the collective sense.

Ground and barge transportation accidents would be expected to result in no additional radiological impacts to the population in the vicinity of the accident. Potential radiological impacts from low probability accidents, which vary considerably, would be dependent on the accident conditions (such as the severity of an associated fire) and the weather conditions at the time of an accident. Since shipping accidents could occur at any location along the routes, it is not possible to identify the racial and economic composition of the populations that might be impacted. In general, however, the principal radiological impacts would be limited to the area within a few miles of the accident location and could be expected to impact a broad mixture of the population in the area.

Tables A-4 and A-5 show minority populations and low-income households, respectively, residing in 800-m (0.5-mi) wide corridors on each side of the road, rail, or barge routes from each of the candidate ports of entry to the Idaho National Engineering Laboratory and the Savannah River Site, both of which could receive spent nuclear fuel in the near term. In these tables, a county is called a "surrounding" county if its boundaries lie at least partially within the 800-m (0.5-mi) corridor. Routes used for this analysis are described in Appendix E.

As a general observation, percentages of minority populations residing along ground transportation routes (Column 7 of Table A-4) from candidate ports on the west coast to the Idaho National Engineering Laboratory are noticeably less than those for transportation from candidate east coast ports to the Savannah River Site. In addition, a higher percentage of minority individuals were found to reside along rail transportation routes than along truck transportation routes. The percentages varied from a minimum

Table A-3 Low-Income Households Residing Near the Candidate Ports

Candidate Port	Total Households Residing within 16 km of Port	Low-Income Households Residing within 16 km of Port	% Low-Income Households Residing within 16 km of Port	Households Residing in Surrounding Counties	Low-Income Households Residing in Surrounding Counties	% Low-Income Households Residing in Surrounding Counties	Households Residing in Surrounding State(s)	Low-Income Households Residing in Surrounding State(s)	% Low-Income Households Residing in Surrounding State(s)
Charleston, SC:									
Wando Terminal	85,851	36,904	43.0	149,358	62,552	41.9	1,258,783	545,937	43.4
NWS Terminal	72,765	32,020	44.0	149,358	62,552	41.9	1,258,783	545,937	43.4
Galveston, TX	29,360	16,607	56.6	81,417	34,984	43.0	6,079,341	2,815,886	46.3
Hampton Roads, VA:									
Newport News	126,789	51,055	40.3	336,638	137,129	40.7	2,294,722	937,123	40.8
Norfolk	206,464	90,723	43.9	336,688	137,129	40.7	2,294,722	937,123	40.8
Portsmouth	175,994	75,147	42.7	336,688	137,129	40.7	2,294,722	937,123	40.8
Jacksonville, FL	125,930	61,052	48.5	290,999	125,610	43.2	5,138,360	2,087,579	40.6
MOTSU, NC	3,071	1,166	38.0	20,094	8,455	42.1	2,517,098	1,067,345	42.4
NWS Concord, CA	143,676	58,344	40.6	415,223	167,426	40.3	10,399,700	4,307,948	41.4
Portland, OR	146,047	66,186	45.3	542,696	222,075	40.9	1,105,362	453,038	41.0
Savannah, GA	57,266	28,960	50.6	125,693	52,772	42.0	3,625,358	1,566,725	43.2
Tacoma, WA	198,458	83,101	41.9	843,736	338,779	40.2	1,875,508	736,285	39.3
Wilmington, NC	45,537	19,491	42.8	79,175	33,226	42.0	2,517,098	1,067,345	42.4

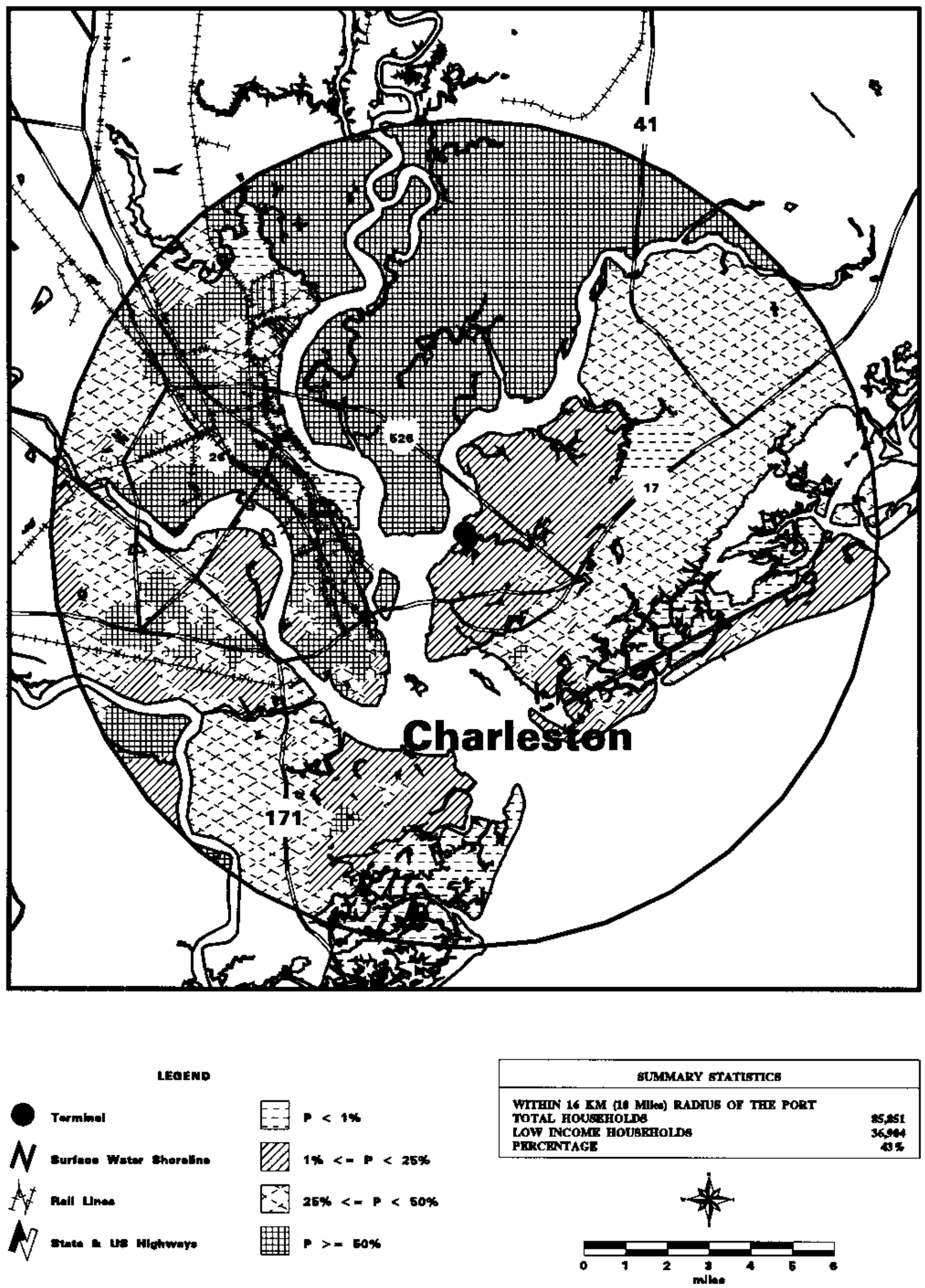


Figure A-12 Distribution of Low-Income Households Residing within 16 km of the Wando Terminal, Port of Charleston, South Carolina

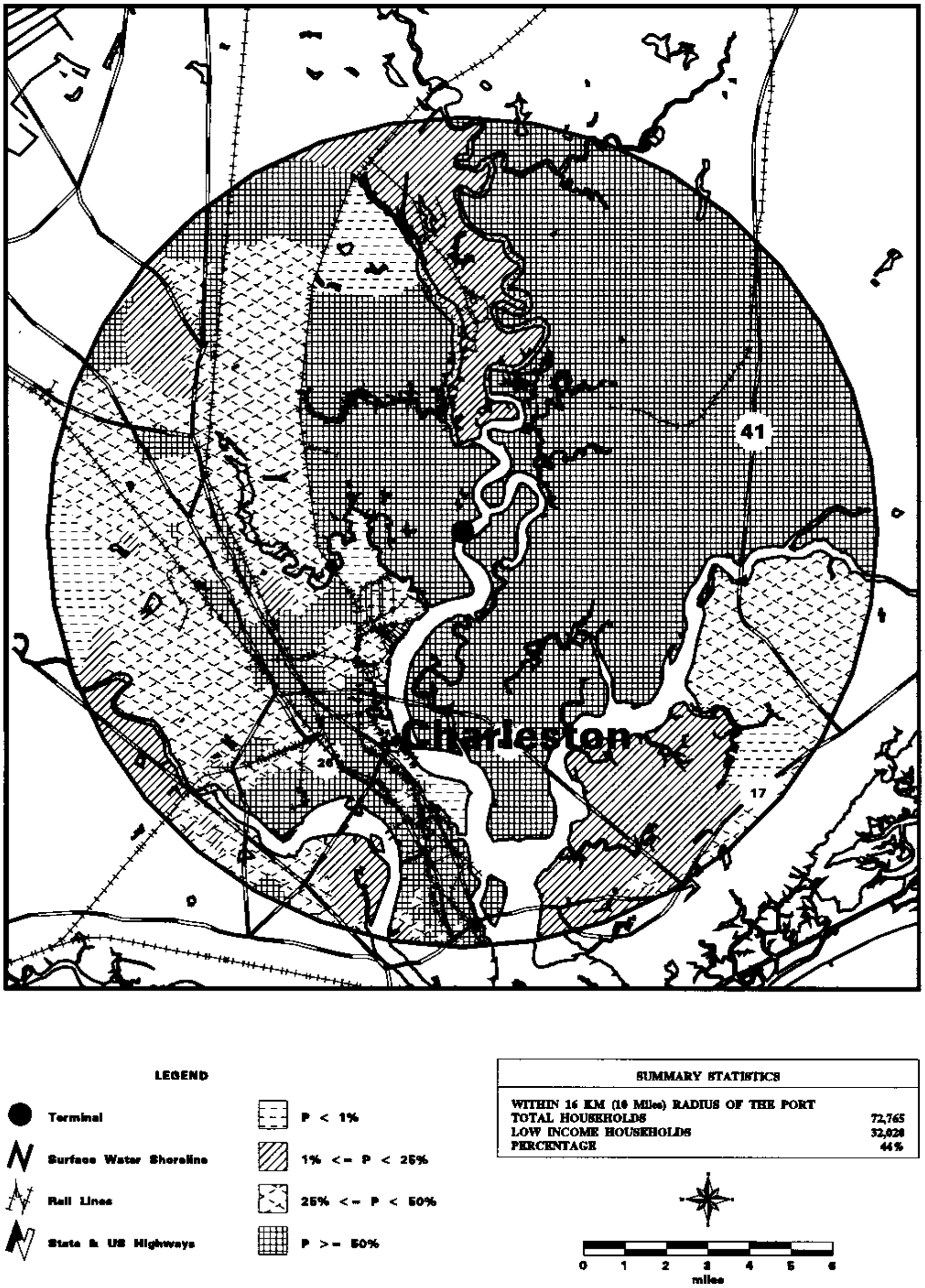


Figure A-13 Distribution of Low-Income Households Residing within 16 km of the NWS Charleston, South Carolina

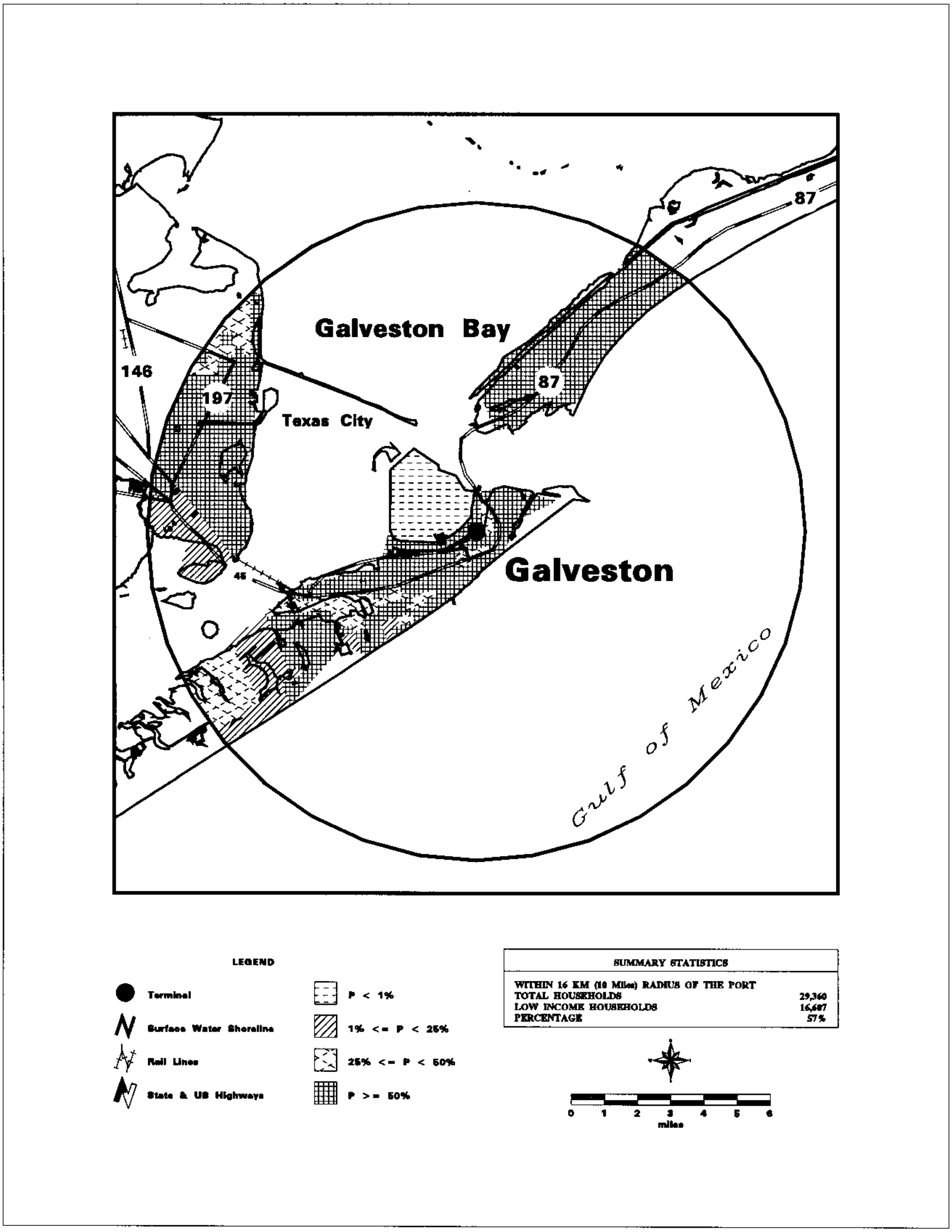


Figure A-14 Distribution of Low-Income Households Residing within 16 km of the Port of Galveston, Texas

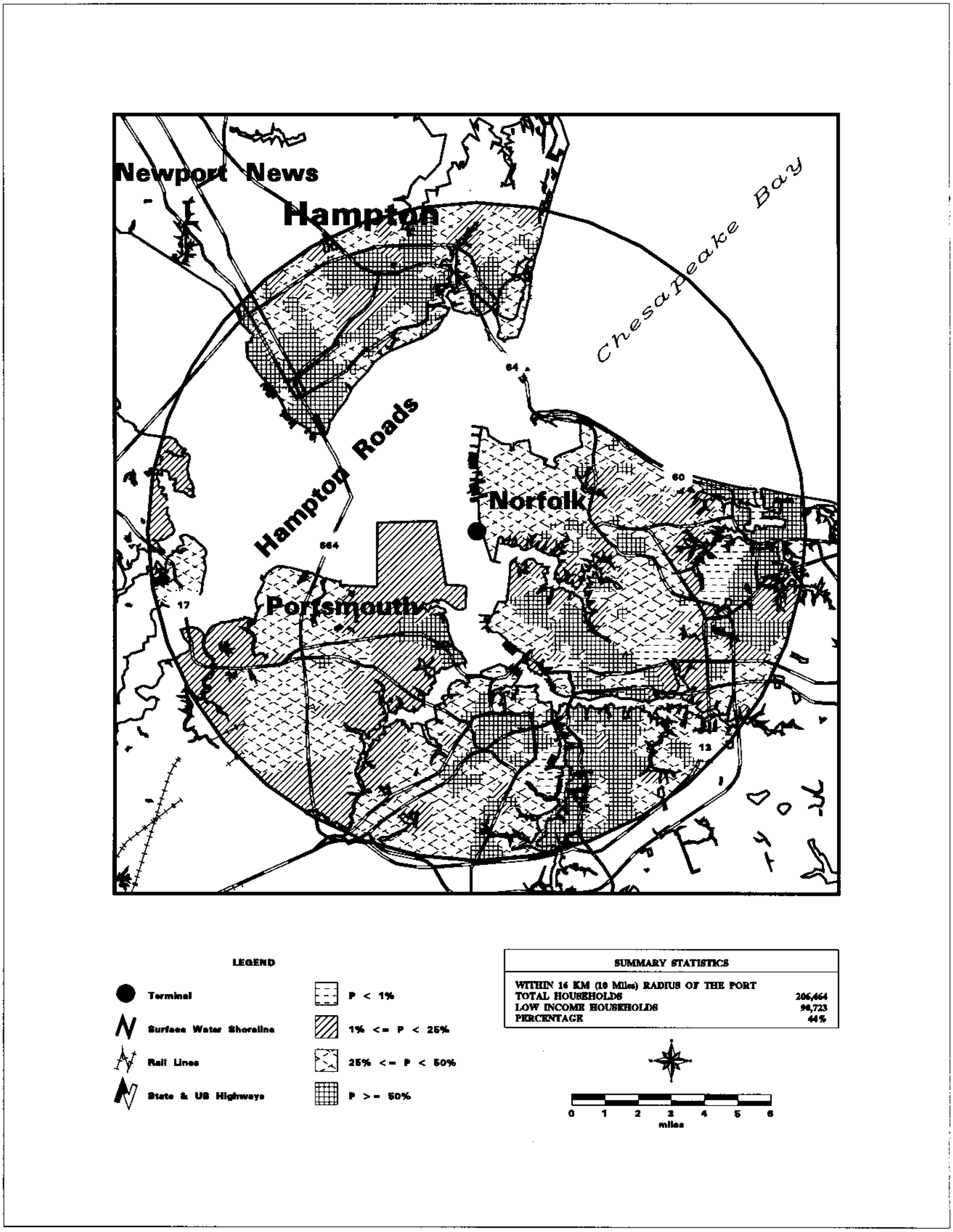


Figure A-15 Distribution of Low-Income Households Residing within 16 km of the Port of Hampton Roads: Newport News, Norfolk, and Portsmouth, Virginia Terminals

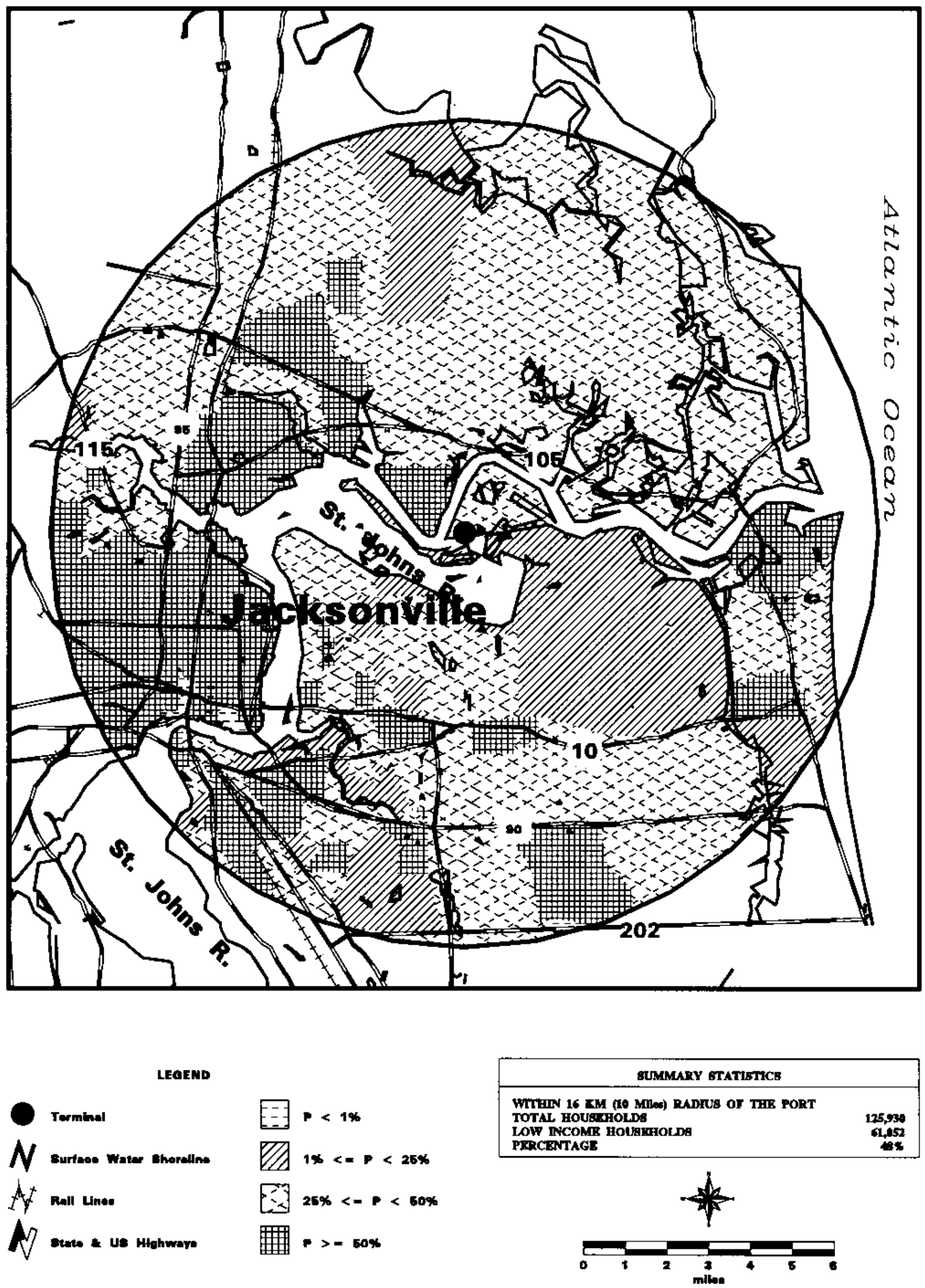


Figure A-16 Distribution of Low-Income Households Residing within 16 km of the Port of Jacksonville, Florida

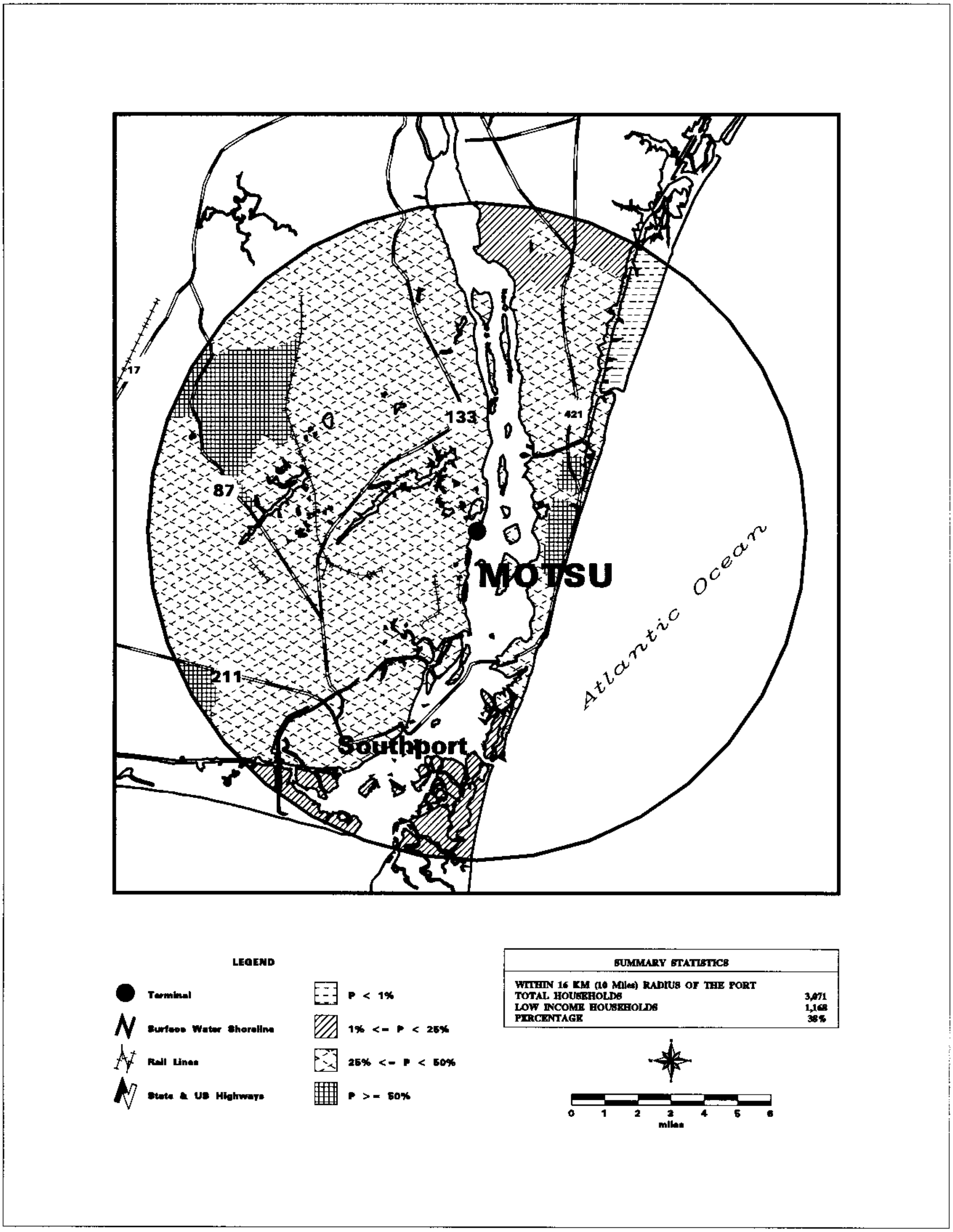


Figure A-17 Distribution of Low-Income Households Residing within 16 km of the Port of MOTSU, North Carolina

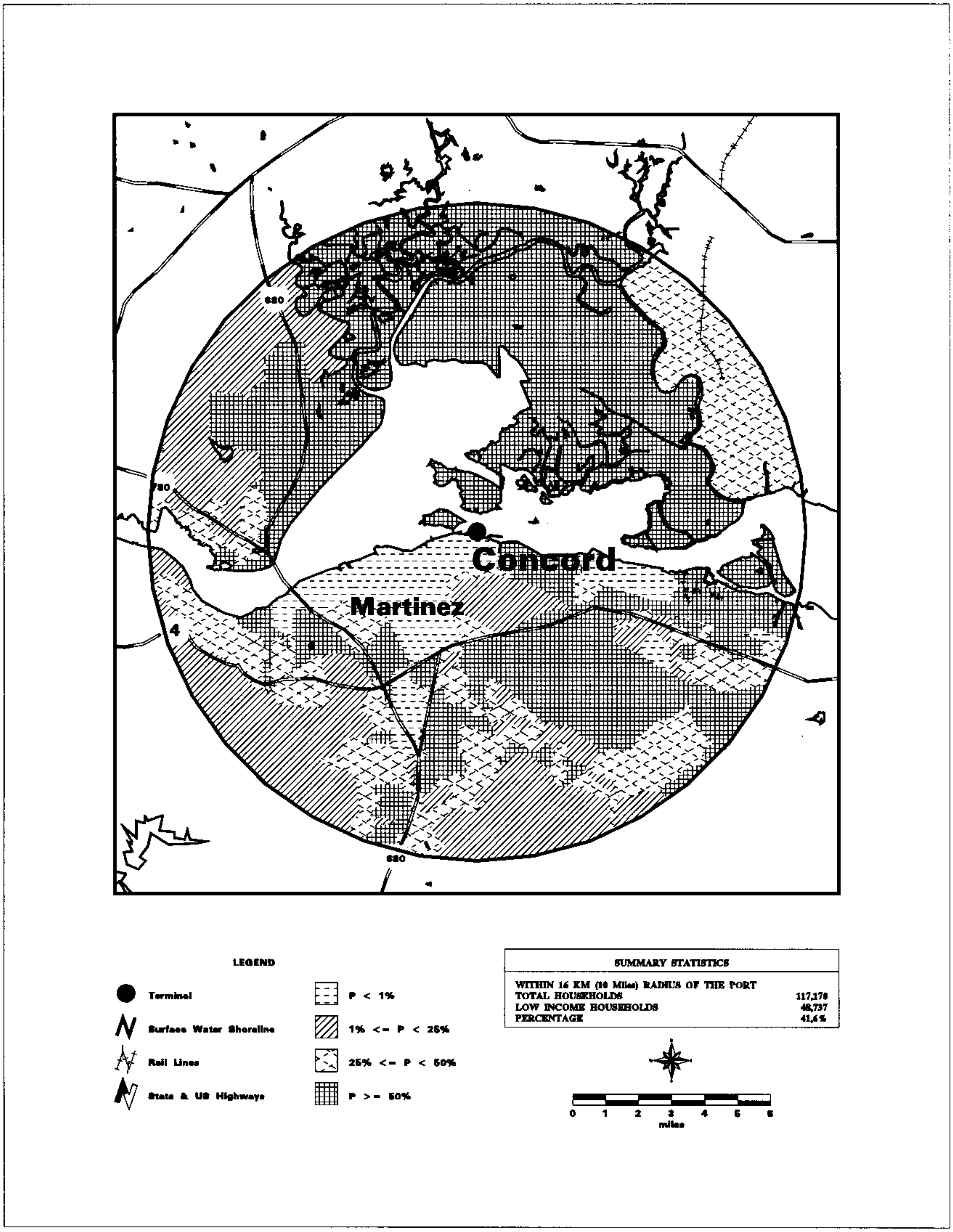


Figure A-18 Distribution of Low-Income Households Residing within 16 km of the Port of NWS Concord, California

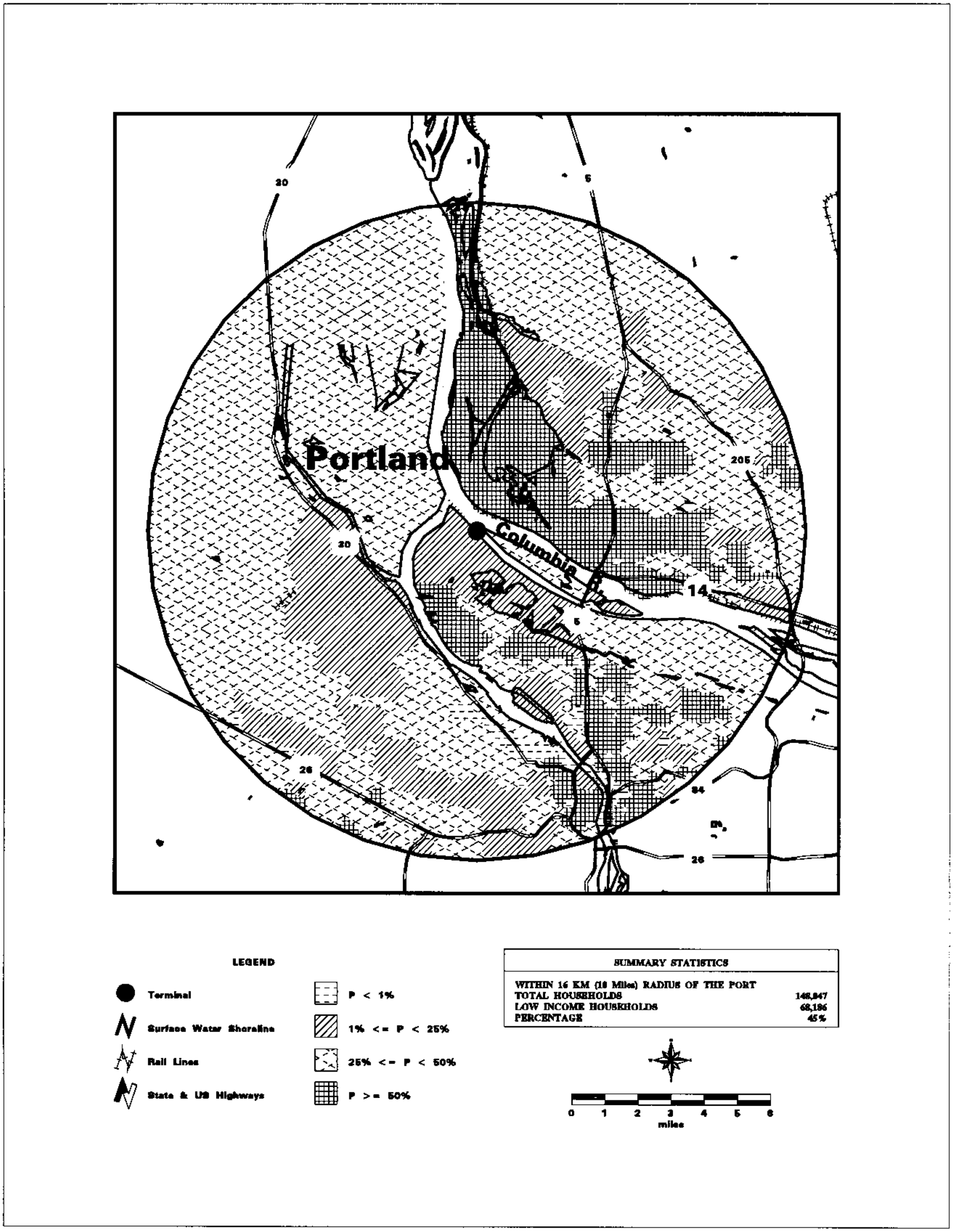


Figure A-19 Distribution of Low-Income Households Residing within 16 km of the Port of Portland, Oregon

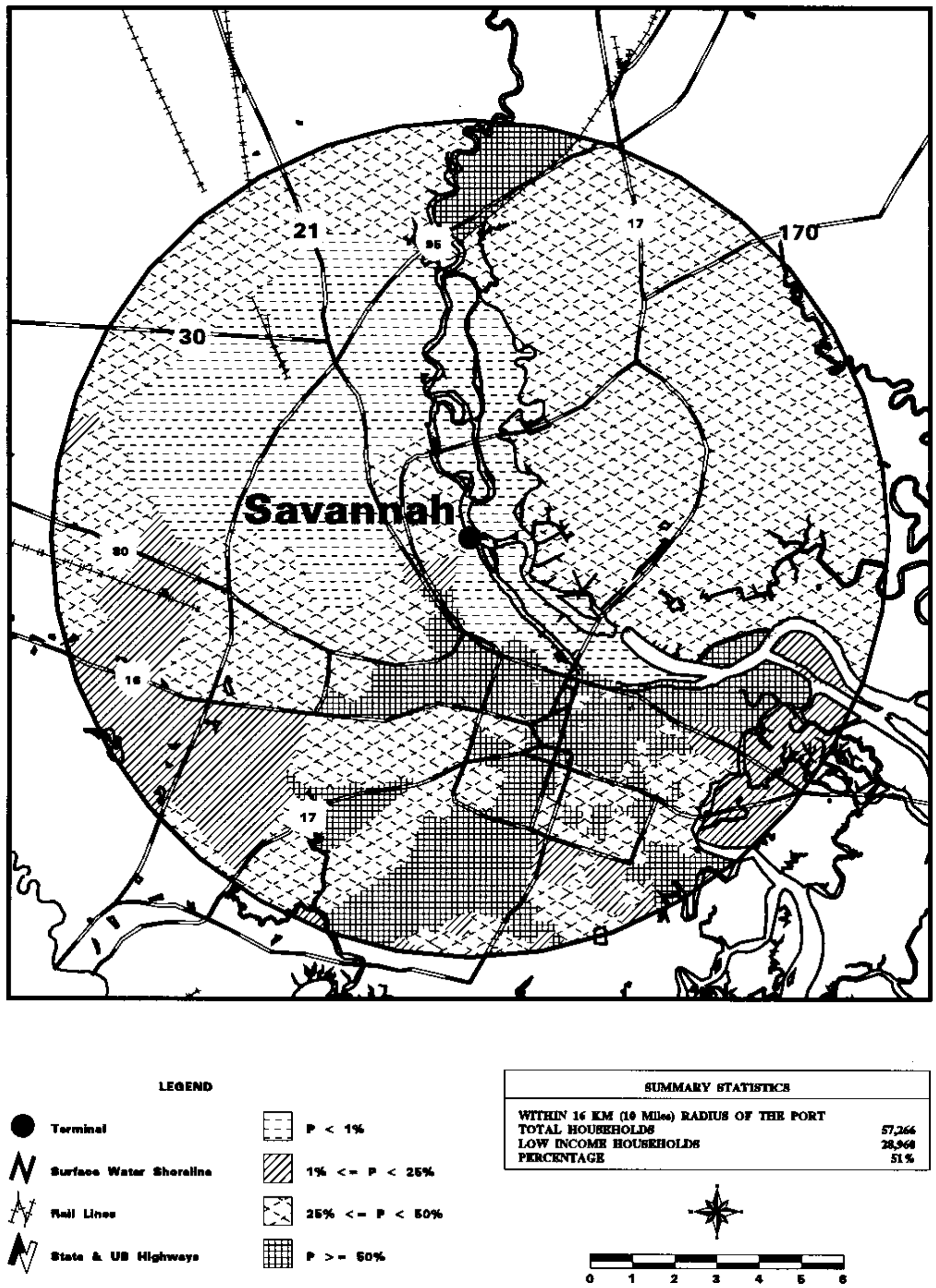


Figure A-20 Distribution of Low-Income Households Residing within 16 km of the Port of Savannah, Georgia

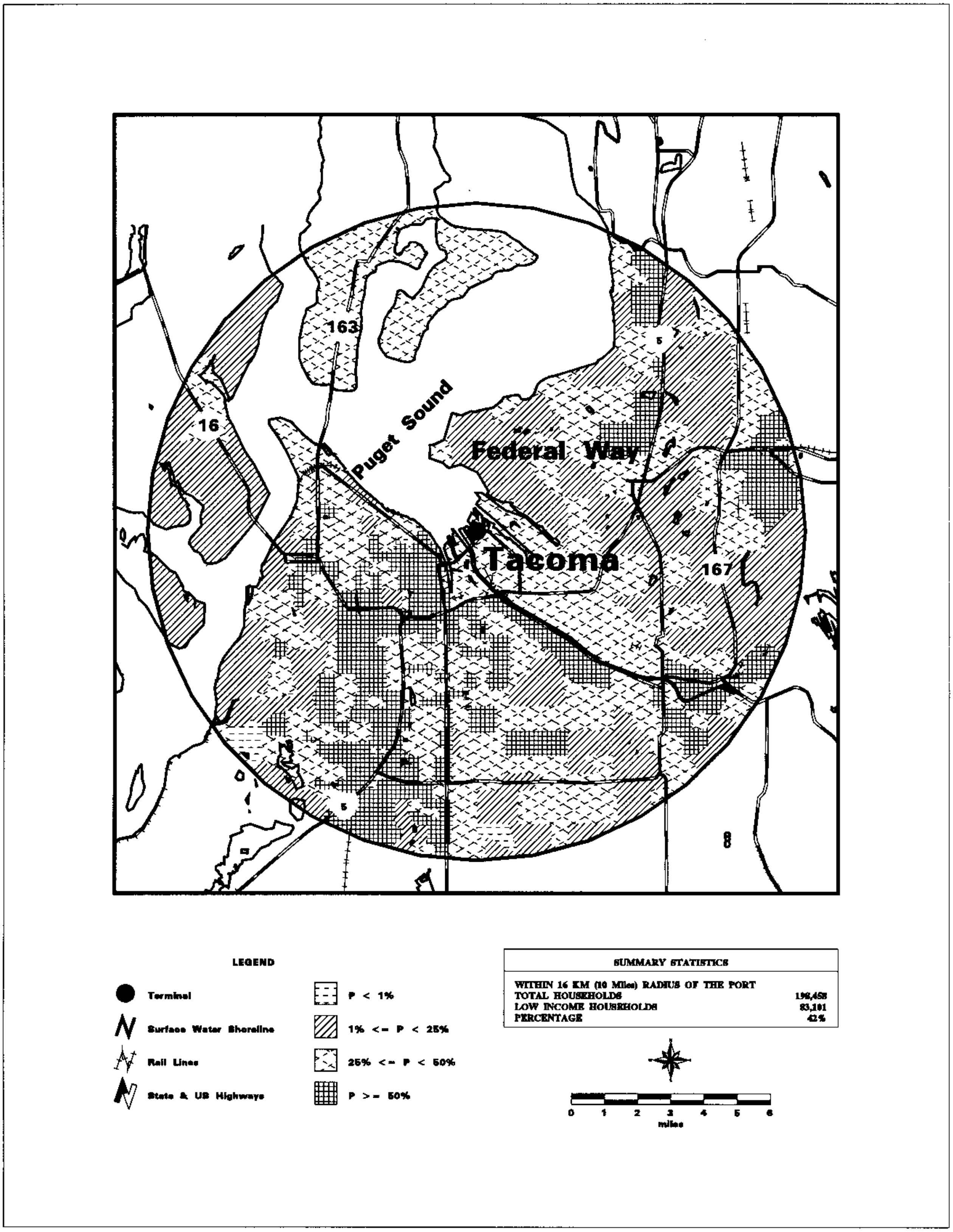


Figure A-21 Distribution of Low-Income Households Residing within 16 km of the Port of Tacoma, Washington

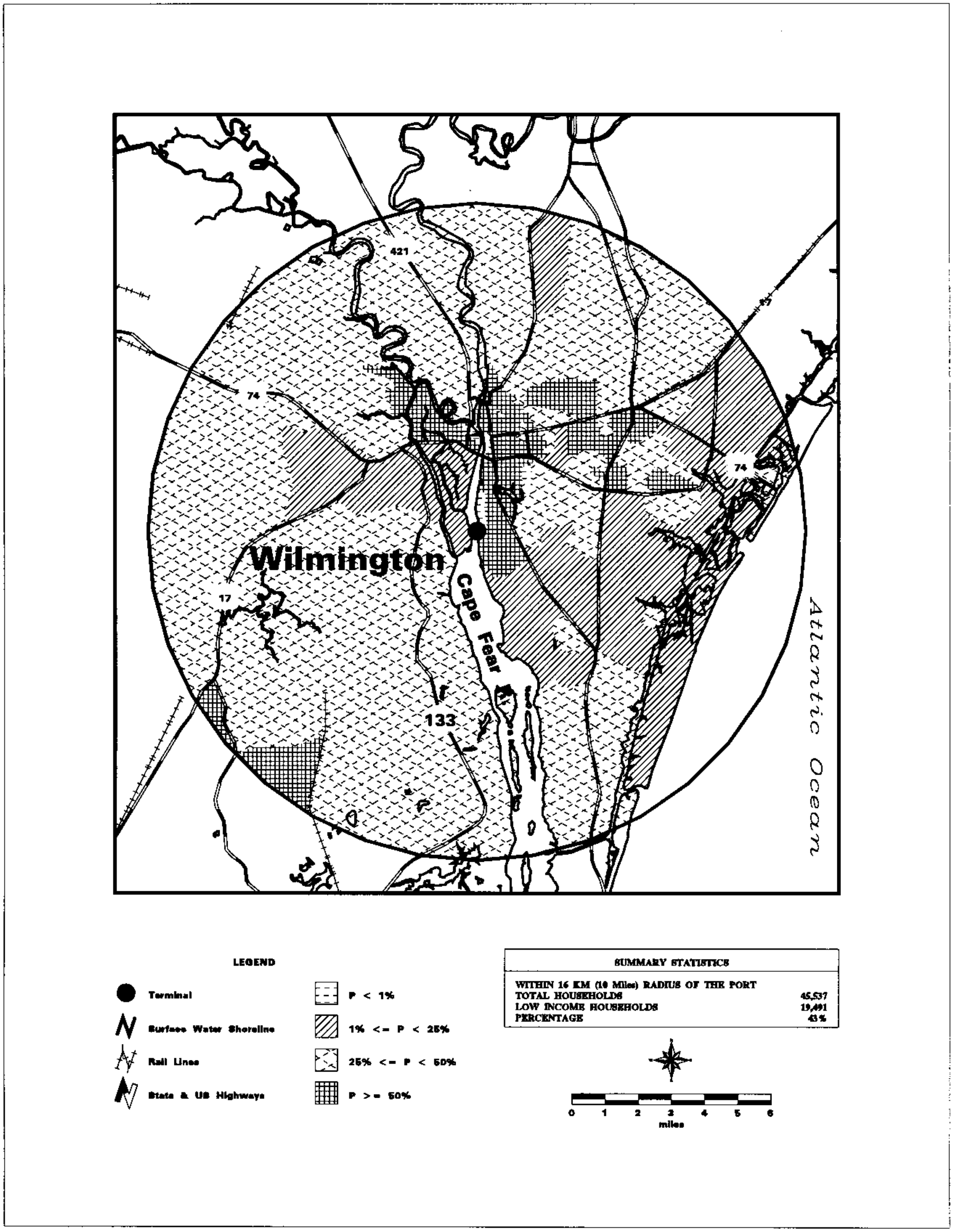


Figure A-22 Distribution of Low-Income Households Residing within 16 km of the Port of Wilmington, North Carolina

Table A-4 Minority Populations Residing Near Ground and Barge Transportation Routes

Route No.	Port	Destination	Mode	Total Pop. Along Route	Minority Pop. Along Route	% Minority Pop. Along Route	Total Pop. in Surrounding Counties	Minority Pop. in Surrounding Counties	% Minority Pop. in Surrounding Counties
	Charleston, SC:								
1	Wando Terminal	INEL	Rail	709,863	177,890	25.1	8,562,589	2,027,753	23.7
2	Wando Terminal	INEL	Truck	514,213	90,978	17.7	8,205,925	1,508,320	18.4
3	Wando Terminal	SRS	Rail	19,633	8,783	44.7	693,370	246,297	35.5
4	Wando Terminal	SRS	Truck	84,729	36,613	43.2	1,180,381	403,989	34.2
5	NWS Terminal	INEL	Rail	709,863	177,890	25.1	8,562,589	2,027,753	23.7
6	NWS Terminal	INEL	Truck	503,081	79,646	15.8	8,205,925	1,508,320	18.4
7	NWS Terminal	SRS	Rail	19,633	8,783	44.7	693,370	246,297	35.5
8	NWS Terminal	SRS	Truck	75,476	31,538	41.8	1,180,381	403,989	34.2
9	Galveston, TX	INEL	Rail	390,876	124,553	31.9	3,794,113	1,006,498	26.5
10	Galveston, TX	INEL	Truck	600,239	205,700	34.3	9,292,668	2,858,758	30.8
11	Galveston, TX	SRS	Rail	528,014	286,872	54.3	8,577,378	3,424,688	39.9
12	Galveston, TX	SRS	Truck	429,057	189,407	44.1	7,965,572	3,038,319	38.1
13	Hampton Roads, VA	INEL	Rail	1,005,972	354,386	35.2	11,758,163	2,982,158	25.4
14	Hampton Roads, VA	INEL	Truck	603,551	115,107	19.1	10,248,206	1,954,088	19.1
15	Hampton Roads, VA	SRS	Rail	221,375	107,613	48.6	2,655,287	984,457	37.1
16	Hampton Roads, VA	SRS	Truck	212,286	98,584	46.4	2,266,251	880,358	38.8
17	Jacksonville, FL	INEL	Rail	697,964	200,596	28.7	8,106,565	1,742,778	21.5
18	Jacksonville, FL	INEL	Truck	622,326	167,802	27.0	10,644,968	2,317,151	21.8
19	Jacksonville, FL	SRS	Rail	52,145	36,707	70.4	1,217,454	388,456	31.9
20	Jacksonville, FL	SRS	Truck	72,821	30,887	42.4	2,154,525	678,172	31.5
21	MOTSU, NC	INEL	Rail	753,535	184,459	24.5	9,084,840	2,087,107	23.0
22	MOTSU, NC	INEL	Truck	515,468	91,325	17.7	9,446,043	1,856,873	19.7
23	MOTSU, NC	SRS	Rail	75,932	33,173	43.7	1,302,260	499,281	38.3
24	MOTSU, NC	SRS	Truck	93,987	40,073	42.6	1,518,891	517,912	34.1
25	NWS Concord, CA	INEL	Rail	344,524	72,137	20.9	4,655,756	973,603	20.9
26	NWS Concord, CA	INEL	Truck	267,109	57,926	21.7	4,655,756	973,603	20.9
27	NWS Concord, CA	SRS	Rail	1,443,296	796,105	55.2	30,242,508	13,433,482	44.4
28	NWS Concord, CA	SRS	Truck	1,240,640	536,731	43.3	28,254,357	11,932,215	42.2

Table A-4 Minority Populations Residing Near Ground and Barge Transportation Routes (Continued)

Route No.	Port	Destination	Mode	Total Pop. Along Route	Minority Pop. Along Route	% Minority Pop. Along Route	Total Pop. in Surrounding Counties	Minority Pop. in Surrounding Counties	% Minority Pop. in Surrounding Counties
29	Portland, OR	HS	Barge	28,430	3,599	12.7	1,099,340	140,872	12.8
30	Portland, OR	INEL	Rail	162,678	26,252	16.1	1,300,552	154,630	11.9
31	Portland, OR	INEL	Truck	124,067	15,463	12.5	1,563,392	172,864	11.1
32	Portland, OR	SRS	Rail	950,116	193,322	20.3	15,663,396	3,695,119	23.6
33	Portland, OR	SRS	Truck	671,113	130,098	19.4	11,149,104	2,395,779	21.5
34	Savannah, GA	INEL	Rail	680,075	178,683	26.3	7,696,157	1,661,208	21.6
35	Savannah, GA	INEL	Truck	574,641	140,293	24.4	9,931,415	2,135,851	21.5
36	Savannah, GA	SRS	Barge	1,715	520	30.3	124,099	56,936	45.9
37	Savannah, GA	SRS	Rail	13,835	11,330	81.9	369,053	151,266	41.0
38	Savannah, GA	SRS	Truck	51,065	18,916	37.0	1,135,620	402,488	35.4
39	Tacoma, WA	INEL	Rail	255,650	45,100	17.6	2,427,856	290,185	12.0
40	Tacoma, WA	INEL	Truck	178,532	29,787	16.7	3,141,728	488,589	15.6
41	Tacoma, WA	SRS	Rail	753,535	184,459	24.5	17,228,536	3,946,748	22.9
42	Tacoma, WA	SRS	Truck	514,955	91,051	17.7	8,850,682	1,470,687	16.6
43	Wilmington, NC	INEL	Rail	753,535	184,459	24.5	9,084,840	2,087,107	23.0
44	Wilmington, NC	INEL	Truck	514,955	91,051	17.7	9,446,043	1,856,873	19.7
45	Wilmington, NC	SRS	Rail	75,932	33,173	43.7	1,302,260	499,281	38.3
46	Wilmington, NC	SRS	Truck	102,951	37,735	36.7	1,518,891	517,912	34.1

INEL = Idaho National Engineering Laboratory, SRS = Savannah River Site, HS = Hanford Site

Table A-5 Low-Income Households Residing Near Ground and Barge Transportation Routes

Route No.	Port	Destination	Mode	Total Households Along Route	Low-Income Households Along Route	% Low-Income Households Along Route	Total Households in Surrounding Counties	Low-Income Households in Surrounding Counties	% Low-Income Households in Surrounding Counties
	Charleston, SC:								
1	Wando Terminal	INEL	Rail	279,468	141,864	50.8	3,256,143	1,359,530	41.8
2	Wando Terminal	INEL	Truck	199,269	81,749	41.0	3,082,221	1,266,095	41.1
3	Wando Terminal	SRS	Rail	7,305	3,358	46.0	242,968	101,094	41.6
4	Wando Terminal	SRS	Truck	31,040	15,664	50.5	419,616	175,993	41.9
5	NWS Terminal	INEL	Rail	279,468	141,864	50.8	3,256,143	1,359,530	41.8
6	NWS Terminal	INEL	Truck	193,945	79,186	40.8	3,082,221	1,266,095	41.1
7	NWS Terminal	SRS	Rail	7,305	3,358	46.0	242,968	101,094	41.6
8	NWS Terminal	SRS	Truck	27,339	13,019	47.6	419,616	175,993	41.9
9	Galveston, TX	INEL	Rail	157,276	83,829	53.3	1,410,581	568,315	40.3
10	Galveston, TX	INEL	Truck	230,042	110,020	47.8	3,471,001	1,440,001	41.5
11	Galveston, TX	SRS	Rail	189,537	103,279	54.5	3,135,368	1,315,512	42.0
12	Galveston, TX	SRS	Truck	157,216	74,900	47.6	2,912,738	1,217,498	41.8
13	Hampton Roads, VA	INEL	Rail	372,127	184,477	49.6	4,350,161	1,774,235	40.8
14	Hampton Roads, VA	INEL	Truck	234,717	98,570	42.0	3,895,065	1,611,870	41.4
15	Hampton Roads, VA	SRS	Rail	83,505	42,094	50.4	963,965	403,155	41.8
16	Hampton Roads, VA	SRS	Truck	80,245	37,427	46.6	782,526	332,806	42.5
17	Jacksonville, FL	INEL	Rail	271,994	140,535	51.7	3,097,768	1,304,381	42.1
18	Jacksonville, FL	INEL	Truck	240,520	106,683	44.4	4,036,596	1,686,985	41.8
19	Jacksonville, FL	SRS	Rail	18,576	10,158	54.7	449,914	193,017	42.9
20	Jacksonville, FL	SRS	Truck	27,436	14,176	51.7	784,169	332,060	42.3
21	MOTSU, NC	INEL	Rail	294,953	148,549	50.4	3,458,175	1,443,995	41.8
22	MOTSU, NC	INEL	Truck	199,145	82,005	41.2	3,591,483	1,507,703	42.0
23	MOTSU, NC	SRS	Rail	28,089	13,791	49.1	465,649	197,699	42.5
24	MOTSU, NC	SRS	Truck	35,670	16,610	46.6	547,984	235,188	42.9
25	NWS Concord, CA	INEL	Rail	128,843	66,118	51.3	1,643,159	675,769	41.1
26	NWS Concord, CA	INEL	Truck	97,241	44,810	46.1	1,643,159	675,769	41.1
27	NWS Concord, CA	SRS	Rail	488,799	257,198	52.6	10,625,645	4,386,618	41.3
28	NWS Concord, CA	SRS	Truck	436,446	184,959	42.4	10,022,988	4,153,610	41.4

Table A-5 Low-Income Households Residing Near Ground and Barge Transportation Routes (Continued)

Route No.	Port	Destination	Mode	Total Households Along Route	Low-Income Households Along Route	% Low-Income Households Along Route	Total Households in Surrounding Counties	Low-Income Households in Surrounding Counties	% Low-Income Households in Surrounding Counties
29	Portland, OR	HS	Barge	11,458	4,582	40.0	428,477	178,142	41.6
30	Portland, OR	INEL	Rail	63,053	29,047	46.1	497,875	206,447	41.5
31	Portland, OR	INEL	Truck	49,120	20,461	41.7	595,780	247,985	41.6
32	Portland, OR	SRS	Rail	368,054	183,842	49.9	5,842,902	2,398,677	41.1
33	Portland, OR	SRS	Truck	261,130	106,936	41.0	4,245,248	1,761,997	41.5
34	Savannah, GA	INEL	Rail	265,122	133,766	50.5	2,932,786	1,231,245	42.0
35	Savannah, GA	INEL	Truck	224,123	97,864	43.7	3,767,606	1,570,456	41.7
36	Savannah, GA	SRS	Barge	595	248	41.7	42,983	18,577	43.2
37	Savannah, GA	SRS	Rail	4,868	2,669	54.8	134,291	56,688	42.2
38	Savannah, GA	SRS	Truck	19,340	9,235	47.8	408,100	171,672	42.1
39	Tacoma, WA	INEL	Rail	98,487	48,094	48.8	917,614	377,168	41.1
40	Tacoma, WA	INEL	Truck	68,303	29,265	42.8	1,212,119	493,235	40.7
41	Tacoma, WA	SRS	Rail	448,663	225,672	50.3	6,455,529	2,633,674	40.8
42	Tacoma, WA	SRS	Truck	223,282	99,101	44.4	3,387,084	1,375,781	40.6
43	Wilmington, NC	INEL	Rail	294,953	148,549	50.4	3,458,175	1,443,995	41.8
44	Wilmington, NC	INEL	Truck	198,884	81,839	41.1	3,591,483	1,507,703	42.0
45	Wilmington, NC	SRS	Rail	28,089	13,791	49.1	465,649	197,699	42.5
46	Wilmington, NC	SRS	Truck	39,759	18,538	46.6	547,984	235,188	42.9

INEL = Idaho National Engineering Laboratory, SRS = Savannah River Site, HS = Hanford Site of 12.5 percent for transportation by truck from Portland, Oregon to Idaho National Engineering Laboratory to a maximum of 81.9 percent for rail transportation from Savannah, Georgia to the Savannah River Site.

As shown in Column 7 of Table A-5, similar observations are true for percentages of low-income households residing along ground transportation routes. In the case of low-income households, percentages varied from a minimum of 41.0 percent for truck transportation from Portland, Oregon and Charleston, South Carolina to the Savannah River Site and Idaho National Engineering Laboratory, respectively, to a maximum of 54.8 percent for rail transportation from Savannah, Georgia to the Savannah River Site.

Populations residing within 1.6 km of barge routes are numerically very small in comparison with those residing near ground transportation routes. Percentages of minority populations and low-income households residing near barge routes are similar to the percentages for ground transportation modes.

A.5 Environmental Justice in Areas Near the Candidate Management Sites

Under normal management site activities associated with receipt and storage of the spent nuclear fuel, the dominant radiological impacts have been shown to be the exposures received by the site workers in the immediate vicinity of the spent nuclear fuel cask. These individuals would be principally those working within the spent nuclear fuel storage facility. The racial and economic composition of these individuals at each management site that would receive the majority of the dose could vary considerably. Health effects due to normal operations and accidents at the five candidate management sites are presented in Section 4.2.4. No latent cancer fatalities or other fatalities would be expected to result from the handling and storage of spent nuclear fuel from foreign research reactors at the sites. At none of the sites would the radiological impacts of either normal releases or low probability accidental releases of spent nuclear fuel be expected to significantly affect the general population outside the management site boundary, including minority and low-income populations. Consequently, there are no adverse impacts of the proposed action on these groups.

A.5.1 Distribution of Minority Populations Near the Candidate Management Sites

The distribution of minority populations residing in various areas surrounding the candidate interim management sites is presented in Table A-6. This table shows minority populations within an 80-km (50-mi) radius centered at the interim management site. For comparison, minority populations are also shown for the counties surrounding each site. A county was included in the analysis if its boundaries lie at least partially within this circle. As shown in the table, minority populations surrounding the Nevada Test Site and the Idaho National Engineering Laboratory are numerically small in comparison with those surrounding the Hanford Site and the Savannah River Site. The minority population surrounding the Nevada Test Site is relatively large because the boundary of the county containing Las Vegas, NV is within 80 km (50 mi) of the site. The Savannah River Site has the largest percentage of minorities in the surrounding area and surrounding counties.

The racial and ethnic composition of minorities surrounding the candidate interim management sites is illustrated in Table A-7. Hispanics composed nearly 81 percent of the minority population surrounding the Hanford Site at the time of the 1990 census. The Hanford Site is also surrounded by a relatively large percentage (about 8 percent) of Native Americans due to the presence of the Yakama Indian Reservation and tribal headquarters in the State of Washington. The area surrounding the Idaho National Engineering Laboratory has the second smallest percentage of minorities of all the sites. The surrounding minority composition is primarily Hispanic, Native American, and Asian. The Fort Hall Indian Reservation lies

Table A-6 Minority Populations Residing Near the Candidate Interim Management Sites

Candidate Management Site	*Population within 80 km of Site*	*Minority Population within 80 km of Site*	*% Minority Population within 80 km of Site*	*Population in Counties Surrounding Site*	*Minority Population in Counties Surrounding Site*	*% Minority Population in Counties Surrounding Site*
Savannah River Site	566,823	214,016	37.8	944,982	330,078	34.9
Idaho National Engineering Laboratory	176,311	15,449	8.8	265,823	21,828	8.2
Hanford Site	383,934	95,042	24.8	565,871	116,610	20.6
Oak Ridge Reservation	863,758	53,185	6.2	1,220,355	65,346	5.4
Nevada Test Site	12,421	2,005	16.1	777,797	186,714	24.0

largely within 80 km (50 mi) of the candidate management site at the Idaho National Engineering Laboratory. Hispanics and African Americans compose nearly 85 percent of the minority population surrounding the Nevada Test Site. The total and minority populations residing within 80 km (50 mi) of the Nevada Test Site are ten times smaller than those of each of the other sites. The Oak Ridge Reservation is surrounded by the smallest percentage of minorities among the five candidate management sites. Minorities residing within 80 km (50 mi) of the site comprise approximately 6 percent of the total population, and African Americans make up nearly 75 percent of this minority population. The Savannah River Site has the largest surrounding minority population of the five candidate interim management sites: African Americans compose approximately 94 percent of the minority population residing within 80 km (50 mi) of this site.

Figures A-23 to A-27 show the distribution of minorities residing within 80 km (50 mi) of each of the candidate management sites. These illustrations were obtained from an analysis of 1990 census data using a geographical information system. The data were obtained from U.S. Bureau of the Census Tiger Line files which contain political boundaries and geographical features, and Summary Tape Files which contain demographic information. Data were resolved to the block group level, usually 250 to 550 household units. In the legend of each figure, "P" denotes the percentage of the total population within block groups comprised of minority members. The most heavily shaded areas shown in these figures indicate block groups for which the minority population exceeds 50 percent.

The minority population residing near the Hanford Site is spread throughout the area with concentrations in directions northeast, southeast, and southwest of the site. By contrast, the minority population surrounding the Idaho National Engineering Laboratory resides in quadrants northeast and southeast of the site. None of the block groups located within 80 km (50 mi) of the Nevada Test Site contained 50 percent of minority residents during the 1990 census. Due to the sparse population surrounding the site, block groups would be relatively large in geographical area. Minorities within 80 km (50 mi) of the Savannah River Site reside throughout the area with concentrations south of the site. As discussed above, no significant radiological health effects are expected for workers or the general population surrounding the five candidate interim management sites, including minority or low income workers.

A.5.2 Distribution of Low-Income Households Near the Candidate Management Sites

Table A-8 demonstrates the number of low-income households in areas surrounding the candidate interim management sites. Except for the Nevada Test Site, the number of low-income households immediately surrounding the sites is typical of the corresponding number for surrounding counties. In the case of the Nevada Test Site, the percentage of low-income households in the area surrounding the site is noticeably larger than that for the relatively affluent nearby counties.

Table A-7 Racial and Ethnic Composition of Minority Populations Residing Within 80 km of Candidate Management Sites

Candidate Manage-ment Sites	*Total Pop.*	*Total Minority Pop.*	*% Minority Pop.*	*Amer. Indian, Eskimo, or Aleut Pop.*	*% Amer. Indian, Eskimo, or Aleut*	*Asian or Pacific Islander Pop.*	*% Asian or Pacific Islander*	*African Amer. Pop.*	*% African Amer.*	*Hispanic Origin Pop.*	*% Hispanic Origin*	*Other Race*	*% Other Race*	*White*	*% White*
SRS	566,823	214,016	37.8	1,136	0.2	5,557	1.0	201,302	35.5	5,874	1.0	145	0.0	352,807	62.2
INEL	176,311	15,449	8.8	3,977	2.3	1,753	1.0	505	0.3	9,075	5.1	138	0.1	160,862	91.2
HS	383,934	95,042	24.8	7,913	2.1	5,296	1.4	4,331	1.1	76,933	20.0	568	0.1	288,891	75.2
ORR	863,758	53,185	6.2	2,985	0.3	4,695	0.5	40,695	4.7	4,518	0.5	290	0.0	810,573	93.8
NTS	12,421	2,005	16.1	273	2.2	49	0.4	634	5.1	1,048	8.4	0	0.0	10,415	83.8

SRS = Savannah River Site, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, ORR = Oak Ridge Reservation, NTS = Nevada Test Site

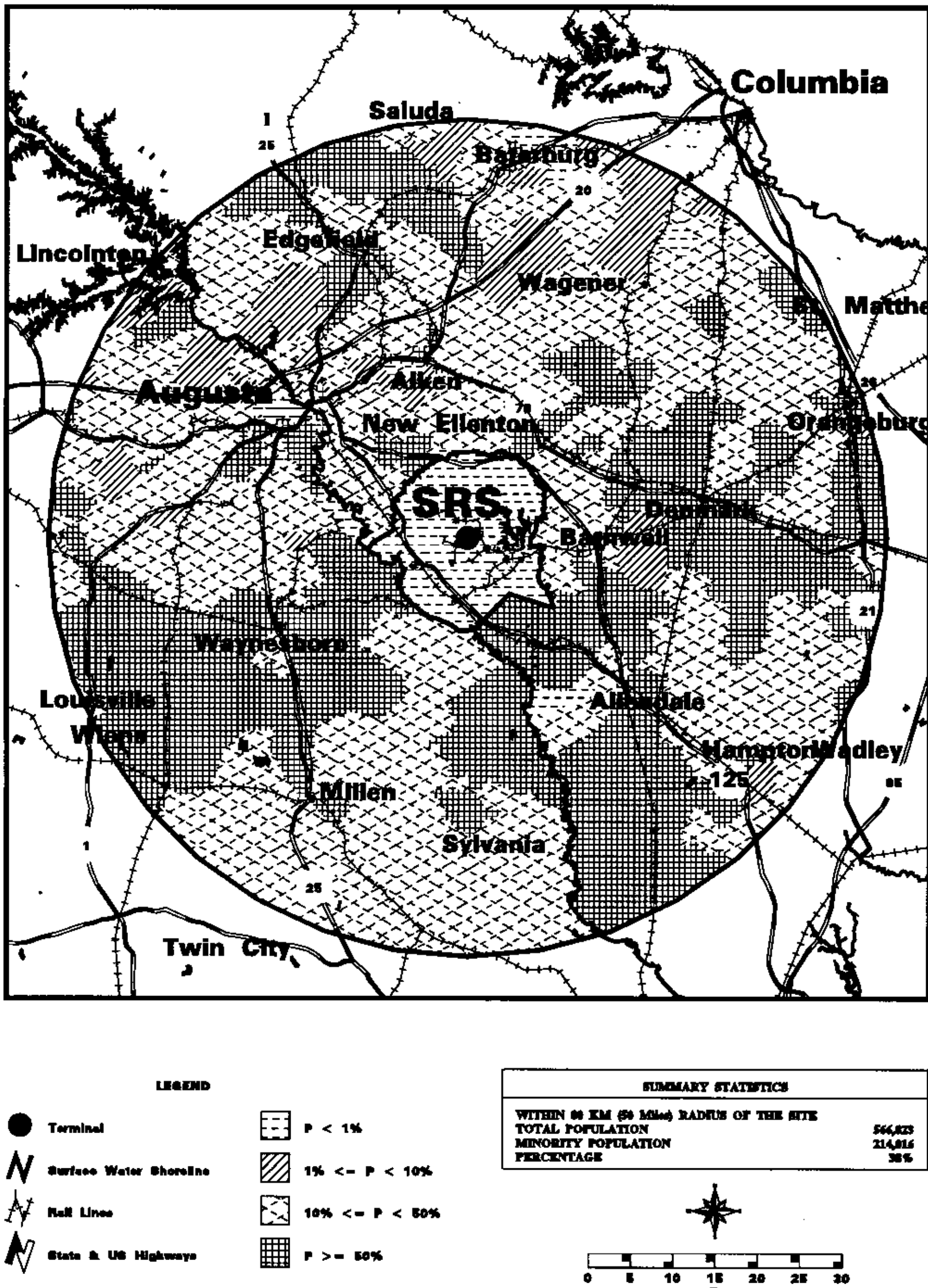


Figure A-23 Distribution of the Minority Population Residing within 80 km of the Savannah River Site

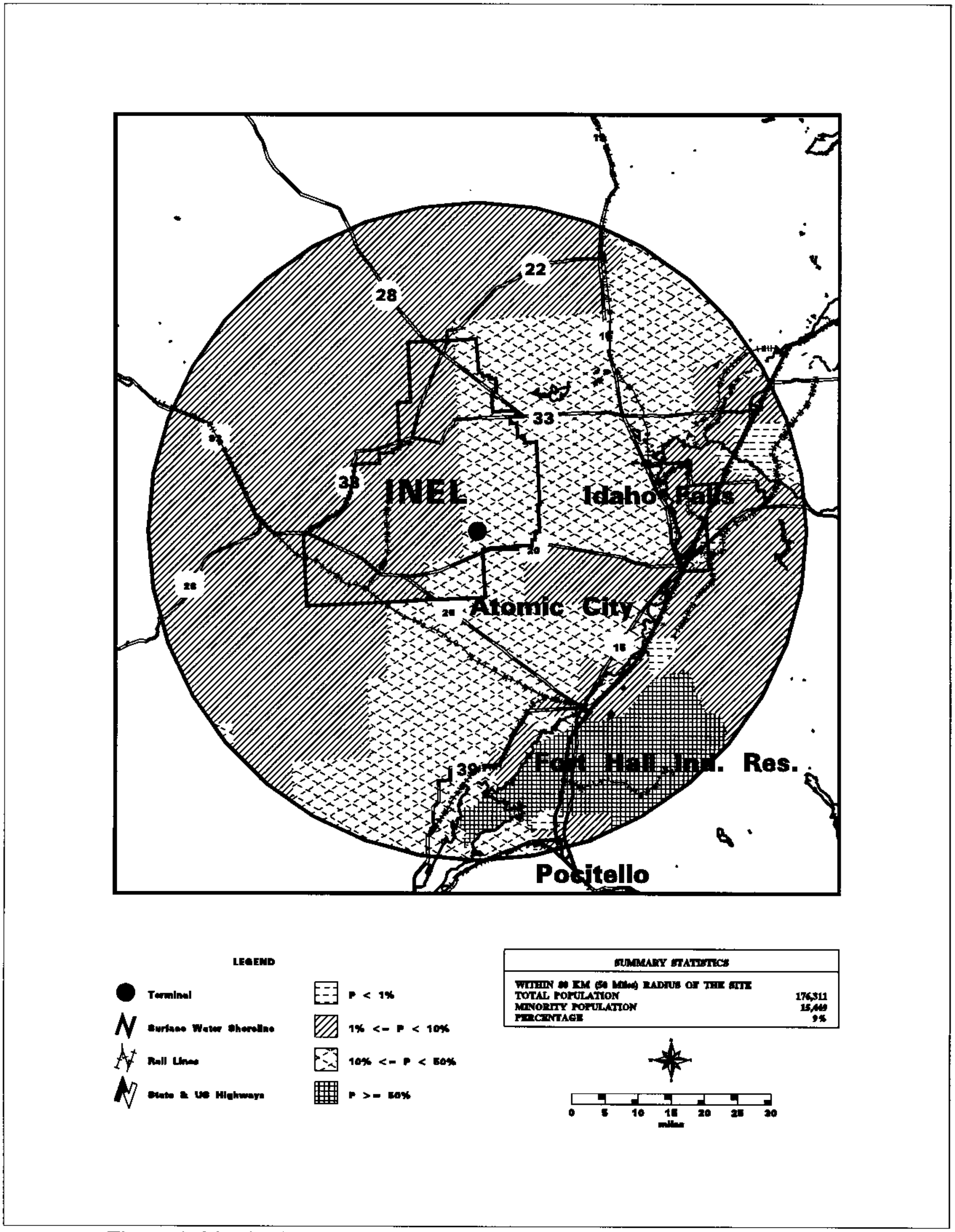


Figure A-24 Distribution of the Minority Population Residing within 80 km of the Idaho National Engineering Laboratory

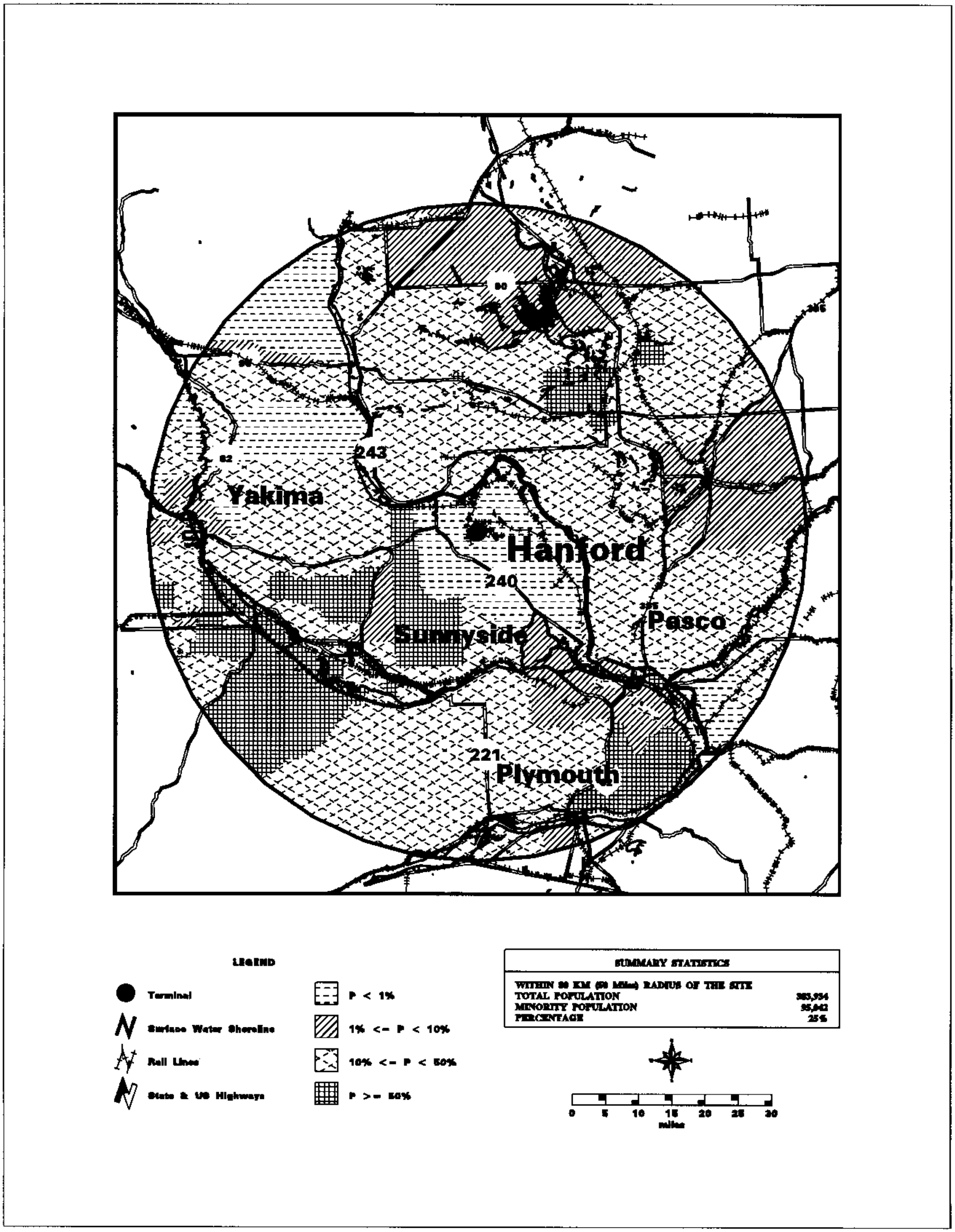


Figure A-25 Distribution of the Minority Population Residing within 80 km of the Hanford Site

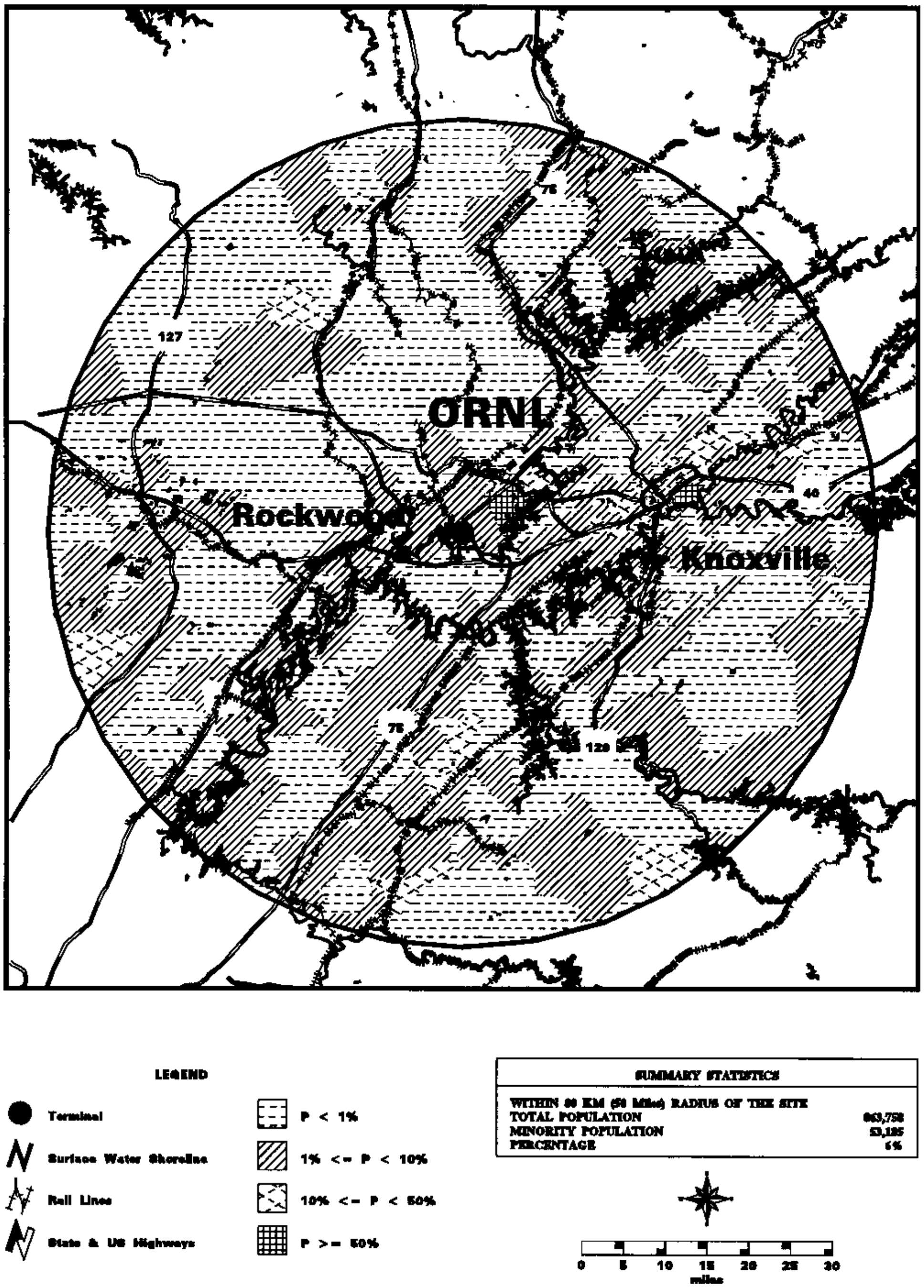


Figure A-26 Distribution of the Minority Population Residing within 80 km of the Oak Ridge Reservation

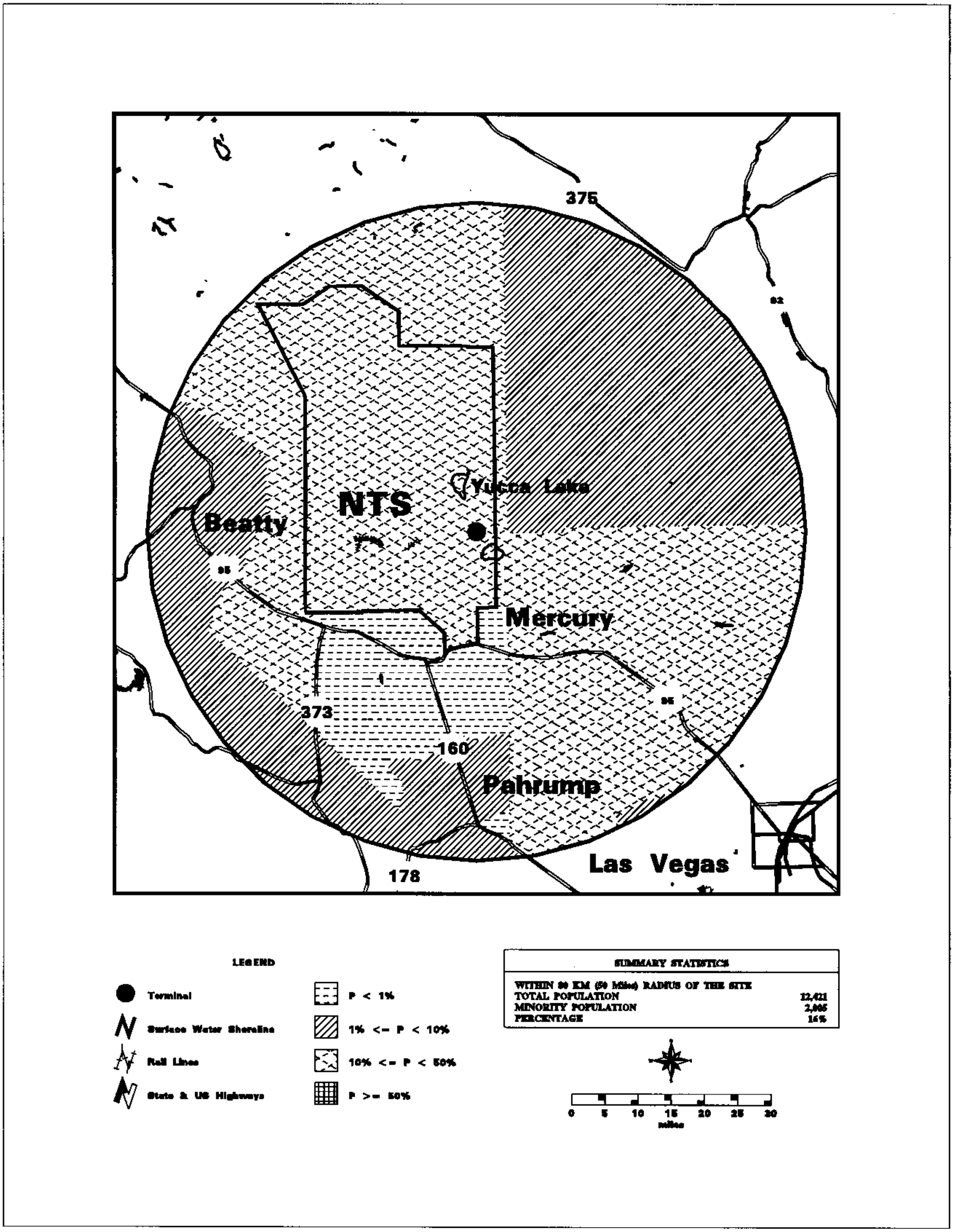


Figure A-27 Distribution of the Minority Population Residing within 80 km of the Nevada Test Site

Table A-8 Low-Income Households Near the Candidate Interim Management Sites

Candidate Management Site	*Households within 80 km of Site*	*Low-Income Households within 80 km of Site*	*% Low-Income Households within 80 km of Site*	*Households in Counties Surrounding Site*	*Low-Income Households in Counties Surrounding Site*	*% Low-Income Households in Counties Surrounding Site*
Savannah River Site	197,937	82,930	41.9	332,193	137,883	41.5
Idaho National Engineering Laboratory	55,109	22,452	40.7	87,723	36,821	42.0
Hanford Site	136,496	57,667	42.2	204,501	86,693	42.4
Oak Ridge Reservation	335,589	147,537	44.0	468,276	206,898	44.2
Nevada Test Site	4,194	2,024	48.3	301,810	119,625	39.6

Figures A-28 through A-32 show the distribution of low-income households within 80 km (50 mi) of each of the candidate interim management sites. The symbol "P" in each legend represents the percentage of low-income households. The heaviest shading indicates where these households total 50 percent or more.

For the Hanford Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, block groups containing 50 percent or more low-income households lie largely south of the site. Low-income households reside throughout the 80-km (50-mi) radius, centered at the Savannah River Site. For the proposed action, no disproportionately high adverse effects are projected for low-income households in the vicinity of the interim management sites.

Characterization of minority and low-income populations residing within a geographical area is sensitive to the basic definitions and assumptions used in conducting the analysis to identify them. Both the Interagency Working Group and DOE are in the process of preparing final guidelines for use in the evaluation of environmental justice. In the absence of final guidance, the definitions and approaches being used by and within Federal agencies could vary. For example, this Final EIS and the Programmatic SNF&INEL EIS present demographic characterizations obtained from the same Census Bureau data base, but use different definitions and assumptions.

The differences in the definitions and assumptions between the Programmatic SNF&INEL EIS and the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Final EIS are as follows:

1. Although both of these EISs use the same 1990 U.S. Census Bureau data base, the Programmatic SNF&INEL EIS uses data aggregated at the census tract level (2,500 to 8,000 persons) while this Final EIS uses data aggregated at the block group level (250 to 550 housing units).

2. In some cases, census blocks or tracts lie partly within the area being analyzed (i.e., within the 80-km (50-mi) radius circle around a potential spent nuclear fuel management site). Since the exact distribution of the populations within such blocks or tracts is not available, the data is insufficient to allow a precise count. To address this situation, the Programmatic SNF&INEL EIS includes a low-income or minority population in its analyses if 50 percent or more of the tract falls within an 80-km (50-mi) radius around the site being considered. In similar situations, this Final EIS assumes that the general population and the minority population are distributed uniformly throughout a block group, and includes the fraction of the low-income or minority population that corresponds to the fraction of the census block group area that falls within the 80-km (50-mi) radius circle.

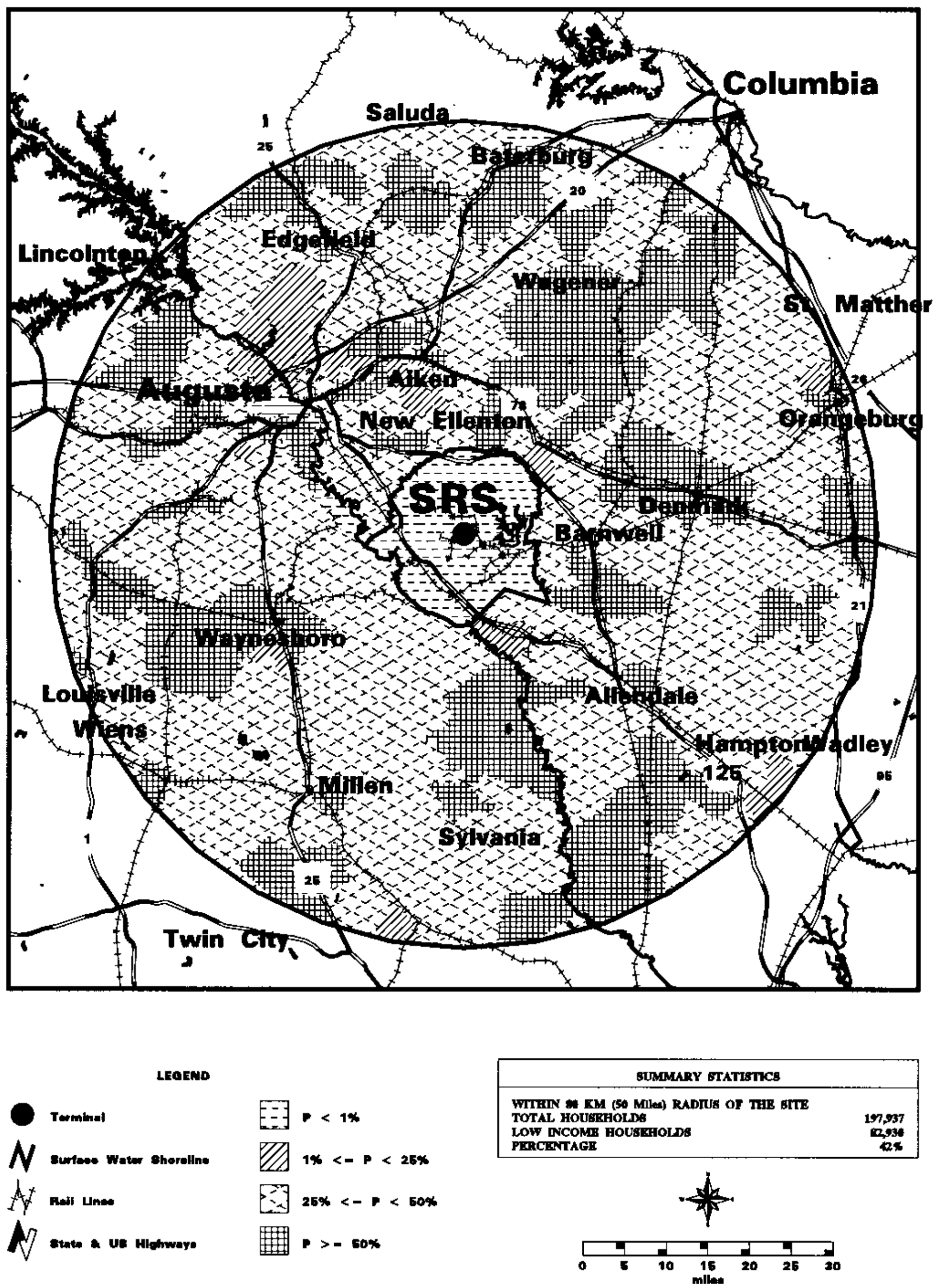


Figure A-28 Distribution of Low-Income Households Residing within 80 km of the Savannah River Site

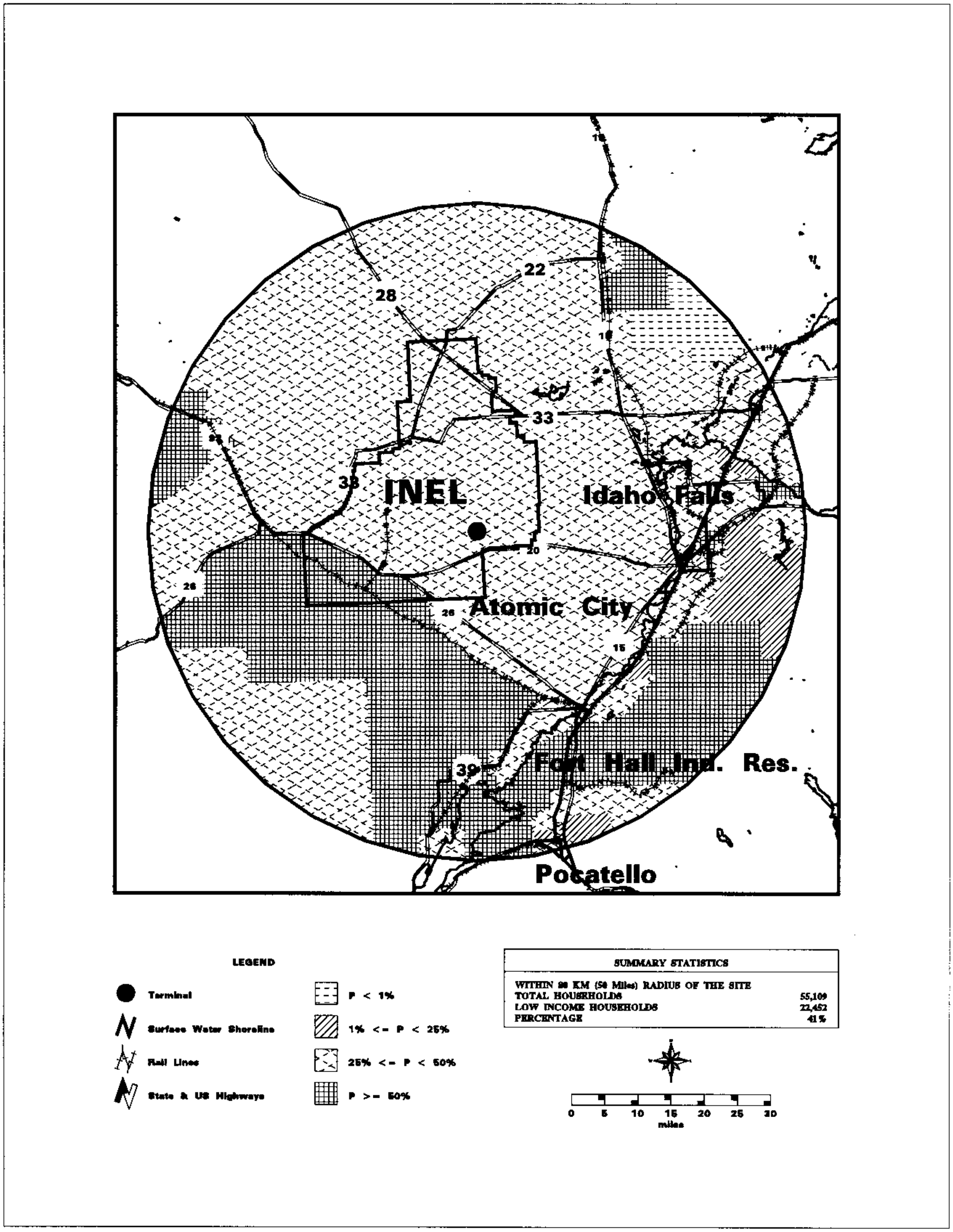


Figure A-29 Distribution of Low-Income Households Residing within 80 km of the Idaho National Engineering Laboratory

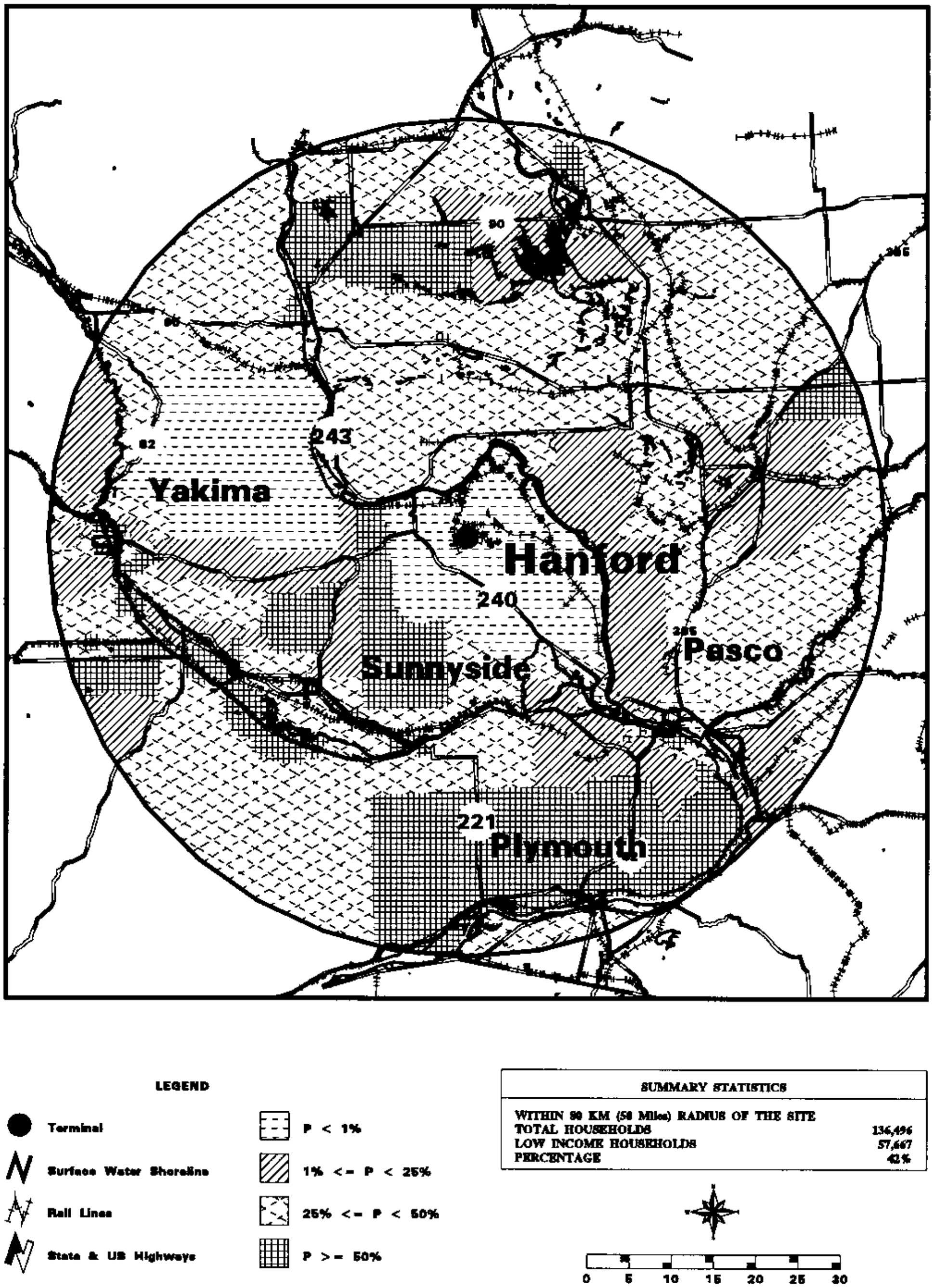


Figure A-30 Distribution of Low-Income Households Residing within 80 km of the Hanford Site

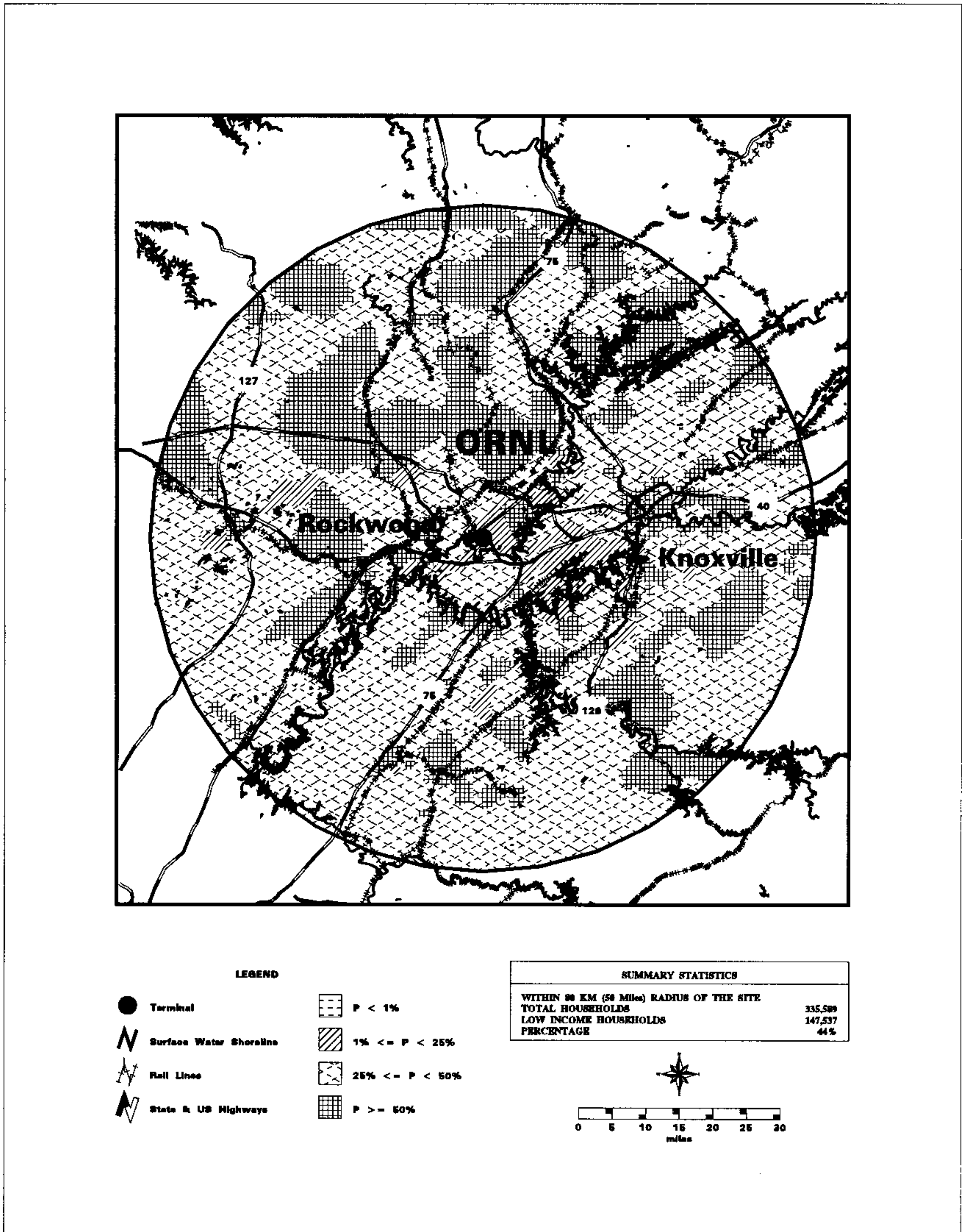


Figure A-31 Distribution of Low-Income Households Residing within 80 km of the Oak Ridge Reservation

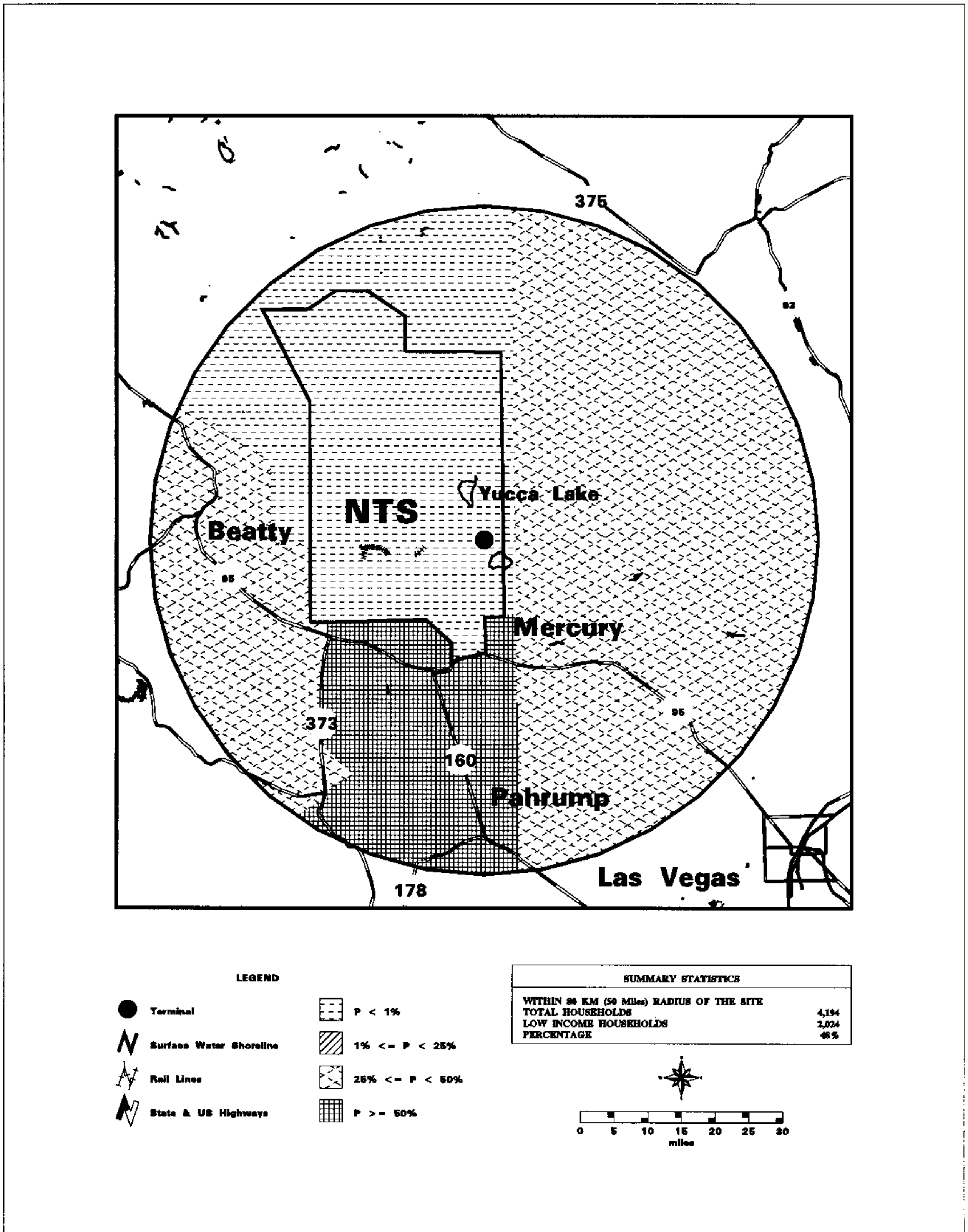


Figure A-32 Distribution of Low-Income Households Residing within 80 km of the Nevada Test Site

3. The Programmatic SNF&INEL EIS defines low-income populations as those in a poverty status as determined annually by the U.S. Census Bureau, based on the Consumer Price Index, and aggregated by the thresholds set forth by the Census Bureau (i.e., a group of people and/or a community experiencing common conditions of exposure or impact, in which 25 percent or more of the population is characterized as living in poverty), a method used by the U.S. Environmental Protection Agency. This Final EIS uses the definition of low-income community established by the U.S. Department of Housing and Urban Development (given in Section A.2 above). Both definitions are permitted under the draft guidance developed by the Interagency Working Group.

These different definitions and assumptions have resulted in differences in the characterization of low-income and minority populations. The two sets of data are summarized in Tables A-9 and A-10 and the most significant differences are discussed below.

Table A-9 Comparison of the Programmatic SNF&INEL EIS's and the FRR SNF Final EIS's Minority Characterization Results

Candidate Management Site	Total Individuals Residing within 80 km (50 mi)		Minority Individuals Residing within 80 km (50 mi)		% of Minority Individuals Residing within 80 km (50 mi)	
	Programmatic SNF&INEL EIS	FRR SNF Final EIS	Programmatic SNF&INEL EIS	FRR SNF Final EIS	Programmatic SNF&INEL EIS	FRR SNF Final EIS
Savannah River Site	619,959	566,823	233,955	214,016	37.7	37.8
Idaho National Engineering Laboratory	172,366	176,311	11,722	15,449	6.8	8.8
Hanford Site	370,807	383,934	75,381	95,042	20.3	24.8
Oak Ridge Reservation	867,231	863,758	49,742	53,185	5.7	6.2
Nevada Tests Site	11,918	12,421	759	2,005	6.4	16.1

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

Table A-10 Comparison of the Programmatic SNF&INEL EIS's and the FRR SNF Final EIS's Low-Income Characterization Results

Candidate Management Site	Total Population Residing within 80 km (50 mi)		Low-Income Group Residing within 80 km (50 mi)		% of Low-Income Group Residing within 80 km (50 mi)	
	Programmatic SNF&INEL EIS (Individuals)	FRR SNF Final EIS (Households)	Programmatic SNF&INEL EIS (Individuals)	FRR SNF Final EIS (Households)	Programmatic SNF&INEL EIS	FRR SNF Final EIS
Savannah River Site	619,959	197,937	107,764	82,930	17.4	41.9
Idaho National Engineering Laboratory	172,366	55,109	23,416	22,452	13.6	40.7
Hanford Site	370,807	136,496	65,584	57,667	17.7	42.2
Oak Ridge Reservation	867,231	335,589	134,661	147,537	15.5	44.0
Nevada Tests Site	11,918	4,194	1,474	2,024	12.4	48.3

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

The minority populations identified are reasonably consistent between the Programmatic SNF&INEL EIS and the FRR SNF Final EIS, except for results obtained at the Nevada Test Site (the largest proportional difference) and the Hanford Site (the largest difference in numbers of individuals), as shown in Table A-9. The range in results for both locations is due to the different aggregations of the demographic data used (census tracts vs. blocks), and the differences in the methods used to account for the population of tracts or groups lying only partly within the area being analyzed, as discussed above. For example, both sites are located in rural or sparsely populated regions so that census tracts surrounding the sites are relatively large in geographical area. In addition, the outskirts of Las Vegas, Nevada begin approximately 80 km (50 mi) from the Nevada Test Site, making the analysis particularly sensitive to differences in treatment of census tracts or block groups that lie partly within a circle of an 80-km (50-mi) radius centered at that site. Most areas within the zone of impact of the Nevada Test Site are restricted access and unpopulated land.

As a result of the different definitions used for identification of low-income populations, the results of these analyses are markedly different, as shown in Table A-10. Both sets of data are correct. They simply reflect the fact that different definitions and assumptions can result in different characterizations of low-income populations.

The approach to evaluating environmental justice used in this document may change as a result of future guidance issued by the Interagency Working Group or DOE. Nevertheless, as demonstrated by the different approaches discussed above, the conclusions are not expected to change because the impacts resulting from the proposed action under all alternatives present no significant risk to the potentially affected populations. As a result, no disproportionately high and adverse effects would be expected for any particular segment of the population, including minority and low-income populations.

References

DOC (U.S. Department of Commerce, U.S. Bureau of the Census), 1992, *1990 Census of Population and Housing, Summary Tape File 3 on CD-ROM*, Washington, DC, May.

DOE (U.S. Department of Energy), 1995, *Programmatic Spent Nuclear Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, Idaho Operations Office, April.

DOE (U.S. Department of Energy), 1994, *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EA-0912, April.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix B
Foreign Research Reactor Spent Nuclear Fuel Characteristics and Transportation Casks

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

List of Tables

Appendix B
Foreign Research Reactor Spent Nuclear Fuel Characteristics and Transportation Casks

B.1 Spent Nuclear Fuel Characteristics

This section presents relevant characterization and other information on foreign research reactor spent nuclear fuel that could be managed under the proposed action. The information includes:

- Estimated amounts of spent nuclear fuel;
- A list of research reactors and foreign countries from which the spent nuclear fuel would originate;
- A description of fuel type design along with important characteristics regarding fuel design, geometry, and burnup;
- A description of the radionuclide inventories for the bounding spent nuclear fuel type; and
- An estimation of the number of foreign research reactor spent nuclear fuel shipments.

B.1.1 Estimated Amount of Spent Nuclear Fuel

The proposed action is for the U.S. Department of Energy (DOE) and Department of State to adopt a policy to manage foreign research reactor spent nuclear fuel which contains uranium enriched in the United States in a manner consistent with the goals of the U.S. nuclear weapons nonproliferation policy (see Chapter 2). The amount of spent nuclear fuel from foreign research reactors that would be managed during the policy period (1995-2005) is approximately 19.2 metric tons of heavy metal (MTHM) with a volume of approximately 110 m^3 (4,100 ft^3) representing approximately 22,700 elements[1] (see Tables B-1 and B-2). Tables B-1 and B-2 provide an estimate of the total amount of spent nuclear fuel that is currently stored or could be generated in each country by late 2005 (Matos, 1994). These tables also provide the estimated number of shipments expected from each country. The number of shipments is a key parameter in evaluating the potential risks associated with the handling and transportation of foreign research reactor spent nuclear fuel (see Section B.1.6). It should be noted that the number of spent nuclear fuel elements and the number of shipments presented for each country in this appendix are estimates based on projections of the numbers of elements to be discharged from foreign research reactors in each country listed over a 10-year period into the future. These estimates are intended to conservatively bound the total number of foreign research reactor spent nuclear fuel elements and shipments associated with the proposed policy. However, the actual distribution of elements and shipments among the listed countries might change, within the limits of the total numbers of elements and shipments listed, based on actual experience gained during the lifetime of any policy that may be established.

---

1 *Various fuel forms and geometries are used in the foreign research reactors (see Section B.1.3). In order to reduce confusion, each individual spent nuclear fuel is called a spent nuclear fuel "element." An element could be an assembly, a rod, a pin, or a cluster of rods or pins.*

Table B-1 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2006

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium (kg)*[b]	*Estimated Number of Shipments*
Argentina[a]	283	71	9
Australia	975	427	9
Austria	157	191	5
Belgium	1,766	730	59
Brazil[a]	155	99	5
Canada	2,831	4,478	116
Chile[a]	58	12	2
Colombia[a]	16	2	1
Denmark	660	529	22
France	1,962	3,442	149
Germany	1,504	909	49
Greece[a]	239	113	8
Indonesia[a]	198	236	6
Iran[a]	29	6	1
Israel	192	111	6
Italy	150	43	5
Jamaica[a]	2	1	1
Japan	2,981	3,134	99
Korea (South)[a]	168	321	7
Netherlands	1,488	1,404	49
Pakistan[a]	82	16	3
Peru[a]	29	39	1
Philippines[a]	50	24	2
Portugal[a]	88	54	3
South Africa[a]	50	10	2
Spain (from Scotland)[c]	40	16	1
Sweden	1,113	1,374	37
Switzerland	159	128	5
Taiwan	127	66	4
Thailand[a]	31	5	1
Turkey[a]	69	89	2
United Kingdom	12	4	1
Uruguay[a]	19	18	1
Venezuela[a]	120	82	4
Total	**17,803**	**18,184**	**675**

[a] *Countries other than high-income economies (World Bank, 1994). These are considered to be "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

[c] *40 spent nuclear fuel elements of Spain's JEN-1 Reactor core are stored in Dounreay, Scotland.*

In addition, in this Environmental Impact Statement (EIS), DOE is considering potential management of highly-enriched uranium (HEU) and low enriched uranium (LEU) target materials from three countries: Canada, Belgium, and Indonesia. These countries have used, and will be using, target fuels which contain U.S.-origin enriched uranium to produce molybdenum-99 (^{99}Mo), which decays to technetium-99 (^{99}Tc), a medical isotope. The amount of target materials that is expected to be brought back to the United States would contain about 556 kg of uranium in 56 to 140 shipments (see Section B.1.5 for detail).

Table B-2 Estimated Number of TRIGA Reactor Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2006

Country	Estimated Number of Spent Nuclear Fuel Elements	Initial Mass of Uranium (kg)[b]	Estimated Number of Shipments
Austria	106	20	3
Bangladesh[a]	100	49	3
Brazil[a]	75	14	3
Finland	171	33	6
Germany	358	68	12
Indonesia[a]	245	47	8
Italy	386	72	13
Japan	326	62	11
Korea (South)[a]	336	64	11
Malaysia[a]	94	47	3
Mexico[a]	186	35	6
Philippines[a]	128	79	4
Romania[a]	1,451	189	48
Slovenia[a]	393	75	13
Taiwan	144	86	5
Thailand[a]	136	35	4
Turkey[a]	79	15	2
United Kingdom	90	17	3
Zaire[a]	136	26	4
Total	**4,940**	**1,033**	**162**

[a] *Countries other than high-income economies (World Bank, 1994). These are identified as "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

The information provided in Tables B-1 and B-2, with regards to the number of spent nuclear fuel elements and the amount of initial mass of uranium, is based on the following assumptions and considerations as compiled by J. Matos of Argonne National Laboratory (Matos, 1994).

B.1.1.1 Fuel Type

Under the "Offsite Fuels Policy" that was in effect during 1988, DOE accepted aluminum-based and Training, Research, Isotope, General Atomic (TRIGA) research reactor fuels[2] for disposition (DOE 1986, and 1987). The "Offsite Fuels Policy" and the current proposed policy pertain to irradiated fuels from foreign nuclear research reactors other than those involved in the conduct of research and development activities leading to demonstration of the practical value of such reactors for industrial or commercial purposes. Specifically, the "Offsite Fuels Policy" and the proposed policy apply solely to the following types of reactor fuels:

2 *Aluminum-based fuel is aluminum-clad and has an active fuel region that consists of an alloy of uranium and aluminum or a dispersion of uranium-bearing compound (e.g., UAl_x, U_3O_8, U_3Si_2, U_3Si) in aluminum. TRIGA fuel consists of an alloy of uranium and zirconium and is clad in either aluminum, incoloy, or stainless steel.*

1. Aluminum-clad reactor fuels where the uranium-235 (^{235}U) content is equal to or greater than 20 percent, by weight, of the total uranium content (i.e., HEU fuel). The active fuel region of these fuels may be configured as uranium-aluminum alloy, uranium-oxide[3] or uranium-aluminide. Spent nuclear fuels containing significant quantities of uranium-233 (^{233}U) are excluded from receipt.

2. Aluminum-clad reactor fuels where the ^{235}U content is less than 20 percent by weight of the total uranium content (i.e., LEU fuel). The active fuel regions of these fuels may be configured as uranium-silicide, uranium-aluminide or uranium-oxide. Fuels containing significant quantities of ^{233}U are excluded from receipt.

3. Aluminum-, incoloy-, or stainless steel-clad, uranium-zirconium hydride (other than ^{233}U) TRIGA fuel types.

In addition to the aluminum-based and TRIGA fuel types discussed above, U.S.-origin enriched uranium is also used in the fuel elements of several fast reactors and other special purpose reactors, in the UO_2 rodded fuel assemblies of several thermal research reactors, and in thermal homogeneous liquid and solid fueled reactors. The enrichment of the uranium ranges from 2 percent to 93 percent. These fuels do not qualify for management under the proposed policy because they were not included in the fuel types that were eligible for return to the United States under the "Offsite Fuels Policy" that was in effect in 1988.

B.1.1.2 Data Sources and Assumptions

Information on current spent nuclear fuel inventories containing U.S.-origin enriched uranium at foreign research reactors and temporary storage facilities was obtained from several sources: (1) questionnaires sent out by DOE and returned by foreign research reactor organizations in 1993 and 1994, (2) data summarized from irradiated fuel questionnaires sent out by and returned to the International Atomic Energy Agency in 1993 and 1994, and (3) Reduced Enrichment for Research and Test Reactors (RERTR) Program information on foreign research reactor fuel inventories, operation, and fuel cycles. Additional information on reactor fuel characteristics and reactor operation was obtained from directories of nuclear research reactors published by the International Atomic Energy Agency (IAEA, 1989).

Beginning with irradiated fuel inventory data, several assumptions were made, first to normalize the data to a common starting date of January 1996, and then to estimate the number of irradiated fuel elements in reactor cores and the number of spent nuclear fuel elements that could be generated during the 10-year policy period (1995-2005). These assumptions are:

1. Most foreign research reactors will continue operation during the 10-year policy period. If a permanent shutdown date has been specified by the research reactor operator, irradiated fuel was accumulated to that date only.

2. The number of irradiated fuel elements in each reactor core was determined from available reports and publications, or estimated. The estimated number of spent nuclear fuel elements covered under the proposed policy includes the inventory within the core of each research reactor at the end of the policy period. This would account for fuel elements in the reactor core of research reactors that shut down during, or at the end of, the policy period.

3 *This uranium-oxide composition refers to aluminum-clad fuel plates or tubes containing dispersions of U_3O_8 in aluminum. It does not include fuels containing UO_2 pellets clad in aluminum, zircaloy, stainless steel, or other materials.*

3. Known current and planned shutdowns for prolonged periods of maintenance and refurbishment have been incorporated into the estimates.

4. Dates for conversion from HEU to LEU fuel have been estimated, and the enrichment change was incorporated into the inventory data.

5. Estimated irradiated fuel inventories have been included for reactors that are under construction and plan to begin operation before the Record of Decision date (assumed here to be December 31, 1995) of the proposed policy using U.S.-origin enriched uranium.

6. Spent nuclear fuel from previously shutdown reactors with fuel in temporary storage has been included.

B.1.1.3 Foreign Research Reactors Eligible for Inclusion in this EIS

There are 104 research and test reactors located in 41 foreign countries that possess aluminum-based and TRIGA fuels containing U.S.-origin enriched uranium. These foreign research reactors are listed in Tables B-3 through B-5. Table B-3 lists 76 reactors that possess aluminum-based fuel only. These foreign research reactors are arranged in a number of categories that depend on each reactor's LEU conversion status. Table B-4 lists 25 foreign research reactors that possess TRIGA fuel only. Table B-5 lists three foreign research reactors that were converted from HEU aluminum-based fuel to LEU TRIGA fuel and thus possess both aluminum-based and TRIGA spent nuclear fuels.

B.1.1.4 Developing Countries

For purposes of this EIS, developing countries are defined as countries having other than high-income economies, on the basis of per capita Gross Domestic Product, by the World Bank (World Bank, 1994). Two countries, Zaire and Taiwan, were not listed in the World Bank report. Zaire is considered here to have a low-income economy; and Taiwan, with an estimated per capita Gross Domestic Product of $10,900 (1994), is considered to have a high-income economy. The countries shown below qualify as developing countries according to this criterion:

List of Developing Countries

Low Income Economies	*Lower Middle Income Economies*		*Upper Middle Income Economies*	
Bangladesh	Chile	Romania	Argentina	Slovenia
Indonesia	Colombia	Thailand	Brazil	South Africa
Pakistan	Iran	Turkey	Greece	South Korea
Zaire	Jamaica		Malaysia	Uruguay
	Peru		Mexico	Venezuela
	Philippines		Portugal	

B.1.2 General Characteristics of Nuclear Fuels and Spent Nuclear Fuel

Nuclear fuels consist of fissile materials that produce a net increase in neutrons when they absorb neutrons, and fertile materials that produce fissile material when they absorb neutrons. The principal fissile materials are ^{235}U, Plutonium-239 (^{239}Pu), and ^{233}U (Plutonium-241 or ^{241}Pu is also of some importance). The principal fertile materials are uranium-238 (^{238}U) and Thorium-232 (^{232}Th) (Plutonium-240 or ^{240}Pu and uranium-234 or ^{234}U also play roles as fertile materials). The only fissile

Table B-3 Foreign Research and Test Reactors that Possess Only Aluminum-Based Fuel Containing HEU and LEU of U.S.-Origin

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin			Comment (see Note)
					Enr.1	Enr.2	Enr.3	
HEU Reactors Fully- or Partially-Converted to LEU Fuel								
1	RA-3	Argentina	3	Plates	90	-	-	(1)
2	ASTRA	Austria	10	Plates	93	45	20	
3	IEA-R1	Brazil	2	Plates	93	-	20	
4	NRU	Canada	125	Pin Cluster	93	-	20	
5	DR-3	Denmark	10	Tubes	93	85	20	
6	OSIRIS	France	70	Plates	-	-	20	
7	FRG-1	Germany	5	Plates	93	-	20	
8	NRCRR	Iran	5	Plates	93	-	-	(2)
9	JMTR	Japan	50	Plates	93	45	20	
10	PARR	Pakistan	5	Plates	92	-	-	(2)
11	R2	Sweden	50	Plates	93	-	20	
HEU Reactors that Have Ordered LEU Fuel Elements for Conversion								
12	GRR-1	Greece	5	Plates	93	-	20	(3)
13	HOR	Netherlands	2	Plates	93	-	20	(3)
14	TR-2	Turkey	5	Plates	93	-	20	(3)
HEU Reactors that Can Be Converted to LEU Fuel								
15	RA-6	Argentina	0.5	Plates	90	-	-	
16	HIFAR	Australia	10	Tubes	80	60	20	(3)
17	SAR-GRAZ	Austria	0.01	Plates	90	-	20	
18	MNR	Canada	2	Plates	93	-	20	
19	Slowpoke - Alberta	Canada	0.02	Pin Bundle	93	-	-	
20	Slowpoke - Halifax	Canada	0.02	Pin Bundle	93	-	-	
21	Slowpoke - Montreal	Canada	0.02	Pin Bundle	93	-	-	
22	Slowpoke - Saskatchewan	Canada	0.02	Pin Bundle	93	-	-	
23	Slowpoke - Toronto	Canada	0.02	Pin Bundle	93	-	-	
24	LA REINA	Chile	5	Plates	80	-	-	
25	IAN-R1	Colombia	0.03	Plates	90	-	-	
26	EOLE	France	0.01	Plates	93	-	-	
27	MINERVE	France	0.003	Plates	93	-	-	
28	SCARABEE	France	20	Plates	93	-	-	
29	Strasbourg - Cronenbourg	France	0.1	Plates	90	-	-	
30	Ulyssee - Saclay	France	0.1	Plates	90	-	-	
31	BER-II	Germany	10	Plates	93	-	20	(3)
32	FRJ-2	Germany	23	Tubes	80	-	20	(3)
33	FRM	Germany	4	Plates	93	45	-	
34	IRR-1	Israel	5	Plates	93	-	20	(3)
35	Slowpoke	Jamaica	0.02	Pin Bundle	93	-	-	
36	JMTRC	Japan	0	Plates	93	45	-	
37	JRR-4	Japan	3.5	Plates	93	-	20	(3)
38	KUCA	Japan	0	Plates	93	45	-	
39	KUR	Japan	5	Plates	93	-	20	(3)
40	UTR Kinki	Japan	0	Plates	90	-	-	
41	HFR Petten	Netherlands	45	Plates	93	-	20	(3)
42	LFR	Netherlands	0.03	Plates	93	-	-	
43	RPI	Portugal	1	Plates	93	-	20	
44	SAFARI	S. Africa	20	Plates	93	-	-	(4)

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin			Comment (see Note)
					Enr.1	Enr.2	Enr.3	
45	R2-0	Sweden	1	Plates	90	-	-	
46	ZPRL	Taiwan	0.01	Plates	93	-	20	
HEU Operating Reactors that Cannot be Converted with Current Technology								
47	BR-2	Belgium	60	Tubes	90-93	-	-	
48	ORPHEE	France	14	Plates	93	-	-	
49	RHF	France	57	Involute Plates	93	-	-	
HEU Operating Reactors Announced to be Shutdown								
50	SILOE	France	35	Plates	93	45	20	
51	SILOETTE	France	0.1	Plates	93	-	-	
52	FMRB	Germany	1	Plates	93	-	-	
53	FRG-2	Germany	15	Plates	90 - 93	-	20	
54	JRR-2	Japan	10	Plates	93	45	-	
55	UTR 300	U. K.	0.3	Plates	90	-	-	
Shutdown Reactors Possessing HEU Fuel								
56	MOATA	Australia	-	Plates	90	-	-	
57	BR-02	Belgium	-	Tubes	90	-	-	
58	NRX	Canada	-	Pin Cluster	93	-	-	
59	PTR	Canada	-	Plates	93	-	-	
60	Slowpoke - Kanata	Canada	-	Pin Bundle	93	-	-	
61	MELUSINE	France	-	Plates	93	-	-	
62	GALILEO	Italy	-	Plates	89	-	-	
63	ISPRA-1	Italy	-	Plates	90	-	-	
64	RANA	Italy	-	Plates	90	-	20	
65	JEN-1	Spain	-	Plates	79	-	20	(5)
66	SAPHIR	Switzerland	-	Plates	93	45	20	
LEU Operating Reactors Possessing Only LEU Fuel								
67	RA-0	Argentina	0.01	Plates	-	-	20	
68	Argonauta	Brazil	0.2	Plates	-	-	20	
69	RSG-GAS30	Indonesia	30	Plates	-	-	20	
70	JRR-3M	Japan	20	Plates	-	-	20	
71	TTR-1	Japan	0.1	Plates	-	-	20	
72	RP-10	Peru	10	Plates	-	-	20	
73	KMRR	S. Korea	30	Pin Cluster	-	-	20	(6)
LEU Shutdown Reactors Possessing Only LEU Fuel								
74	THAR	Taiwan	-	Plates	-	-	20	
75	RU-1	Uruguay	-	Plates	-	-	20	
76	RV-1	Venezuela	-	Plates	-	-	20	

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed or anticipated to be possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Note:

(1) *Converted to LEU fuel of Soviet origin.*

(2) *Converted to LEU fuel of Chinese origin.*

(3) *Use of fuel containing LEU of U.S.-origin is anticipated to begin before 2001.*

(4) *Currently uses HEU of South African origin.*

(5) *JEN-1 fuel is currently being stored in Dounreay, Scotland.*

(6) *The KMRR reactor in South Korea began operation using LEU aluminum-based fuel in January 1995.*

Table B-4 Foreign Research and Test Reactors that Possess Only TRIGA Fuel Containing HEU and LEU of U.S.-Origin

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin		
					Enr.1	Enr.2	Enr.3
Reactors Possessing HEU Fuel							
1	Vienna	Austria	0.25	Rods	70	-	20
2	Salazar	Mexico	1	Rods	70	-	20
3	SSR	Romania	14	Rods	93	-	20
4	Ljubljana	Slovenia	0.25	Rods	70	-	20
5	Seoul #2	S. Korea	2	Rods	70	-	20
Reactors Possessing LEU Fuel							
6	Dhaka	Bangladesh	3	Rods	-	-	20
7	Belo Horiz.	Brazil	-	Rods	-	-	20
8	Helsinki	Finland	0.25	Rods	-	-	20
9	Hannover	Germany	-	Rods	-	-	20
10	Heidelberg	Germany	0.25	Rods	-	-	20
11	Mainz	Germany	0.1	Rods	-	-	20
12	Bandung	Indonesia	1	Rods	-	-	20
13	Yogyakarta	Indonesia	0.1	Rods	-	-	20
14	Pavia	Italy	0.25	Rods	-	-	20
15	Rome	Italy	1	Rods	-	-	20
16	Mushashi Inst	Japan	0.1	Rods	-	-	20
17	NSRR-Tokai	Japan	0.3	Rods	-	-	20
18	Rikkyo U.	Japan	0.1	Rods	-	-	20
19	Kuala Lumpur	Malaysia	1	Rods	-	-	20
20	ACPR	Romania	0.5	Rods	-	-	20
21	Seoul #1	S. Korea	0.25	Rods	-	-	20
22	Istanbul	Turkey	0.25	Rods	-	-	20
23	Imp Chem Ind.	U. K.	0.25	Rods	-	-	20
24	TRICO II	Zaire	1	Rods	-	-	20
Shutdown Reactors							
25	TRICO I	Zaire	-	Rods	-	-	20

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Table B-5 Foreign Research and Test Reactors that Possess Both Aluminum-Based and TRIGA Fuel Containing HEU and LEU of U.S.-Origin.

	Reactor	Country	Power, MW	Fuel Geometry	Initial Enrichments[a] Wt-% ^{235}U, U.S. Origin		
					Enr.1	Enr.2	Enr.3
1	PRR-1	Philippines	3	TRIGA Rods	-	-	20
			-	Plates	93	-	20
2	THOR	Taiwan	1	TRIGA Rods	-	-	20
			-	Plates	93	-	
3	TRR-1	Thailand	2	TRIGA Rods	-	-	20
			-	Plates	90	-	-

[a] *Initial enrichments, in weight-% ^{235}U, of the fuels possessed by each reactor. Only fuels containing uranium of U.S.-origin are included.*

Note:

All three of these reactors have been converted from plate-type, aluminum-based HEU fuel to TRIGA LEU fuel. The PRR-1 reactor in the Philippines possesses both HEU and LEU cores of plate-type aluminum-based fuel elements.

material that occurs in nature in a significant quantity is ^{235}U. Natural uranium consists of 0.711 weight percent (w/o) ^{235}U, 99.283 w/o ^{238}U; and 0.0055 w/o ^{234}U as a negligible trace constituent. Uranium-235 is the only fissile material used in foreign research reactors.

In a research reactor, the fuel matrix typically consists of enriched uranium metal in an alloy of aluminum or zirconium hydride. The enriched uranium may contain up to 93 weight percent ^{235}U. The fuel matrix form is either plates (flat or curved), tubes made of three curved plates, or pellets combined into rods. The cladding is the encapsulation (typically aluminum or stainless steel) that surrounds the fuel for confinement and protection. The structural part of a fuel element holds fuel plates or tubes in the proper configuration and directs coolant flow (light or heavy water) over the fuel. Structural parts are usually aluminum. The fuel rods do not require additional structural parts. The size of a fuel element ranges from approximately 1 kg (2.2 lb) to more than 100 kg (220 lb), and lengths range from 76 to 300 cm (2.5 to 9 ft).

As the fuel in a reactor is irradiated, it undergoes nuclear transmutations that cause its composition to change. In the reactor, the fissionable materials in the fuel undergo a process called "fission reaction." Fission reaction occurs when an atom of ^{235}U interacts with a free neutron causing the ^{235}U atom to split into two lighter nuclei which are referred to as "fission products." The fission reaction also results in the release of heat and additional free neutrons that are available to sustain the fission reaction or to maintain criticality. In addition to fission products, heavier elements such as plutonium and other isotopes of uranium are formed when uranium in the fuel absorbs free neutrons rather than undergoing the fission process. The changes in composition of the fuel bring about changes in the fission reaction rate of the fuel. As the reactor operation continues, the fission reaction rate decreases and eventually the reactor will no longer remain critical unless some spent nuclear fuels are replaced with fresh fuels. The discharged fuel is called "spent nuclear fuel." The extent of change in the composition of the fuel is expressed in terms of "burnup," in either percent (atom percent) of fissile material consumed, or the number of megawatt days of heat released per element (or per metric ton of uranium).

When initially discharged from the reactor, spent nuclear fuel is highly radioactive and generates a significant amount of heat. Therefore, the spent nuclear fuel must be stored in a wet pool that provides both shielding and cooling environments. The cooling is required in order to prevent the spent nuclear fuel from being damaged by the heat that fission products generate, and the shielding is needed to protect the workers who handle the fuel.

The quantity of radioactive material in spent nuclear fuel, and the resulting heat generation, decreases over time because of decay of fission products in the spent nuclear fuel. Radioactive decay refers to a process whereby the radioactive elements undergo nuclear transformations that ultimately convert them to stable (nonradioactive) elements. Many fission products formed during reactor operation have short half-lives (the time required for a quantity of radioactive material to decrease to one-half of its original amount) and others remain radioactive for tens to thousands of years. The high initial quantities of fission products in the spent nuclear fuel put the greatest requirements on providing shielding and cooling during the first few months after the spent nuclear fuel is discharged from the reactor. The rapid decay of short half-lived radioactive material leads to reduction of the amount of radioactive material in the spent nuclear fuel over time. This, in turn, reduces the need for continued storage of the spent nuclear fuel in a wet pool. After about 1 year, the heat generation rate in a spent nuclear fuel element decreases to about one percent of the level present at the time of its discharge from the reactor, and this heat generation rate would not damage the spent nuclear fuel if it is stored in a "dry" cask in preparation for transportation and dry storage.

B.1.3 Foreign Research Reactor Spent Nuclear Fuel Designs

Foreign research reactors use a number of different fuel designs. These designs can be organized into five categories: (1) plate-type design, (2) concentric tube-type design, (3) pin-type design, (4) special-type design, and (5) rod-type design. The first four designs are aluminum-based fuel while the fifth is a TRIGA type. The first two fuel types (plate-and tube-type fuels) are known as material test reactor (MTR) fuels. The following summarizes specific characteristics of the different types of fuel named above.

B.1.3.1 Plate-Type Design

This type of fuel design is used in the majority of foreign research reactors. The thermal power of these reactors ranges from 1 MW to 50 MW. Figures B-1 and B-2 show typical fuel elements using this type of fuel design. The number of fuel plates in an element varies between 6 and 23, and the initial ^{235}U content varies between 37 g (1.3 oz) and 420 g (14.8 oz) per element. Similarly, the average burnup of a discharged spent nuclear fuel varies between 15 and 76 percent (^{235}U atom percent). The uranium enrichment in this type of fuel varies from just below 20 to 93 percent.

The following provides additional information on a typical plate-type spent nuclear fuel element which was used in a 50 MW foreign research reactor, as shown in Figure B-2.

The fuel element is made of an alloy of 23 percent by weight of 93 percent enriched uranium in aluminum with a thin (0.38 mm) aluminum cladding. Each fuel element contains 19 fuel plates. The nominal dimensions and weights of each fuel plate and the fuel element are:

	Fuel Plate	*Element*	*Element (cut)*
Dimensions (mm):			
Length	778	1,200	800
Width	70.8	77.0	77.0
Height	1.27[a]	77.0	77.0
Weight (g):			
^{235}U	15	285	285
Total	202	--	5,500
Burnup:			
^{235}U (g)	~3	60	60

[a] *Thickness*

The cut element reflects that portion of the fuel element that contains fuel material. The aluminum nose cone and the aluminum top section of the fuel element are cut to reduce the size of the spent nuclear fuel prior to shipment. This action is usually performed at the foreign research reactor site if the site is equipped to do so. The cutting is necessary to pack more cut elements in a transportation cask, and also since some casks cannot accommodate the whole element length.

B.1.3.2 Concentric Tube Design

This type of fuel design is used in four foreign research reactors: Australian (HIFAR), Belgian (BR-2), Japanese (JRR-2) and Danish (DR-3). The Belgian reactor is a 125 MW reactor, and the other three are each 10 MW. Figure B-3 shows a typical fuel element using concentric tube (tubular) fuel type. The number of fuel tubes in an element varies between 4 and 6, and the initial ^{235}U content varies between

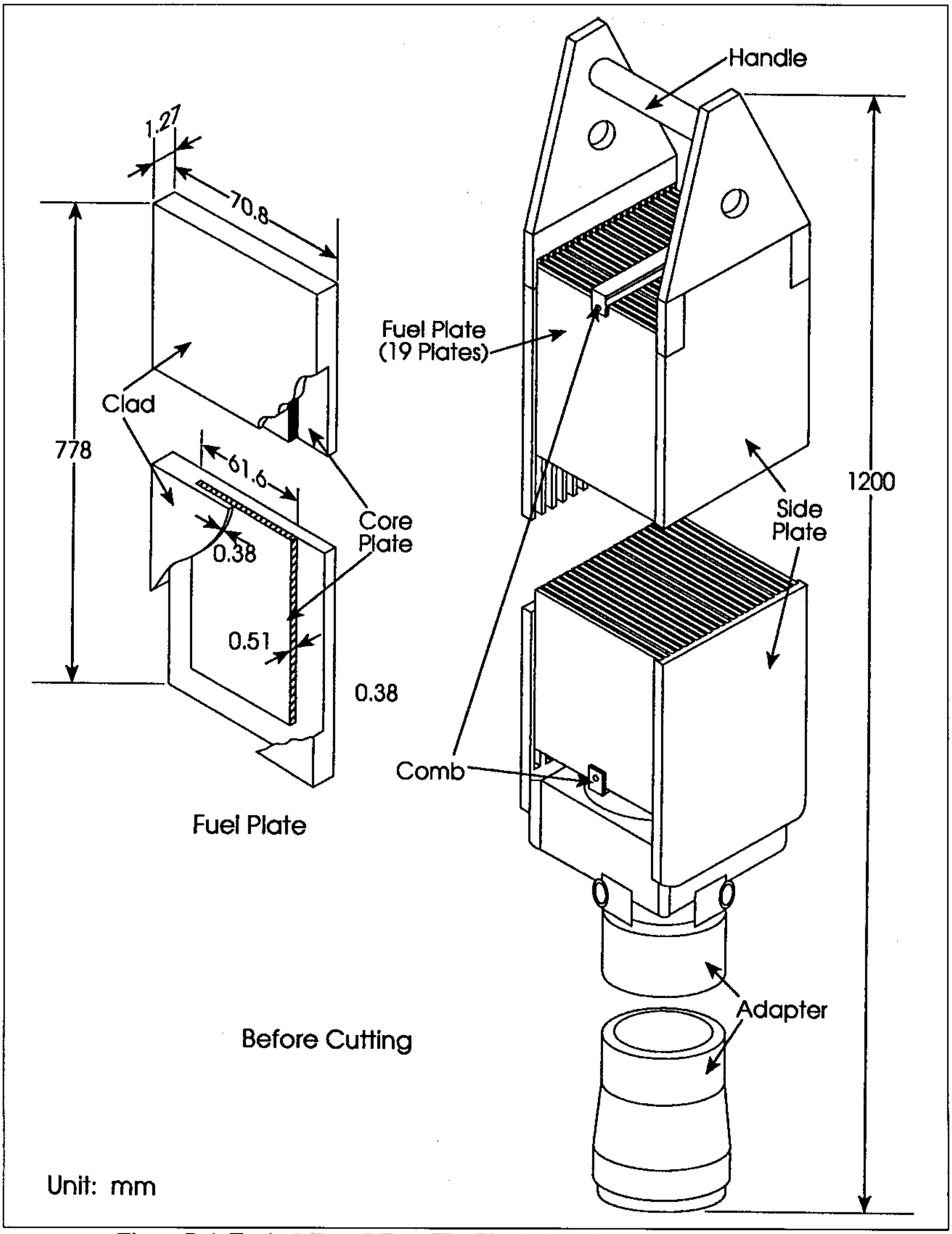


Figure B-1 Typical (Boxed-Type/Flat Plate) Aluminum-Based Fuel Element Schematic

Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3R	Side Plate	Aluminum Alloy	1
3L	Side Plate	Aluminum Alloy	1
4	Outer Fuel Element	U-Al Alloy	2
5	Inner Fuel Element	U-Al Alloy	15
6	Comb	Aluminum Alloy	1
7	Comb Pin	Aluminum Alloy	1

Unit = mm

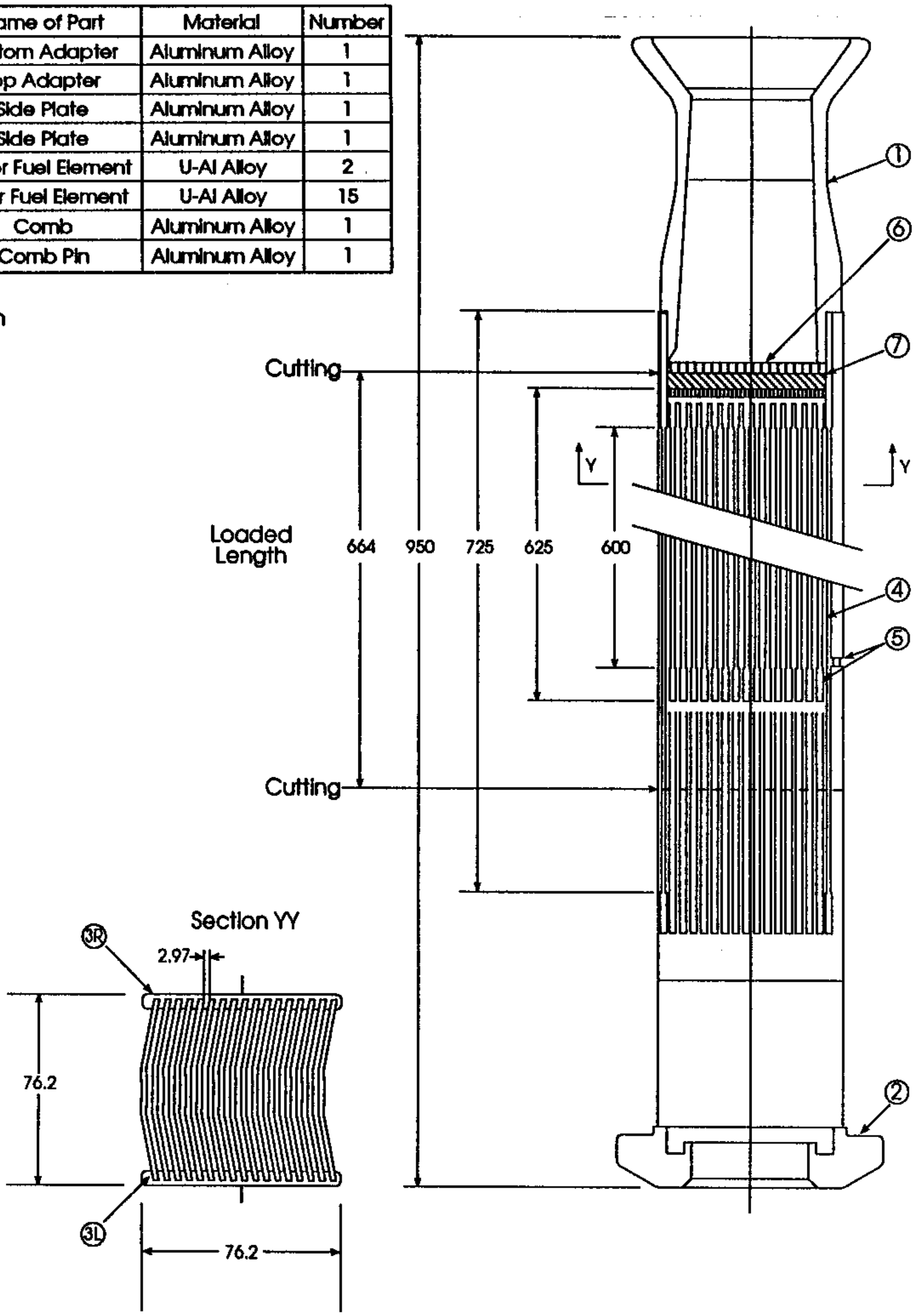
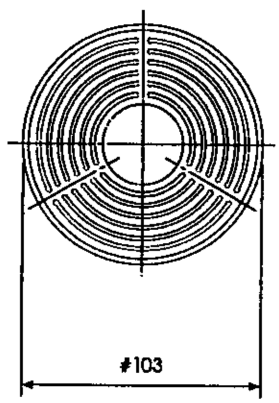


Figure B-2 Typical (Boxed-Type/Curved Plate) Aluminum-Based Fuel Element Schematic

Part No.	Name of Part	Material	Number
1	Bottom Adapter	Aluminum Alloy	1
2	Top Adapter	Aluminum Alloy	1
3	Outer Cylinder	Aluminum Alloy	1
4	Plate	Aluminum Alloy	3
5	Fuel Plate	———	5x3
6	Inner Layer Plate	Aluminum Alloy	3
7	Top Capsule Guide	Aluminum Alloy	1
8	Comb Plate	Aluminum Alloy	3
9	Capsule Guide	Aluminum Alloy	1
10	Comb	Aluminum Alloy	3

Unit = mm



Section AA

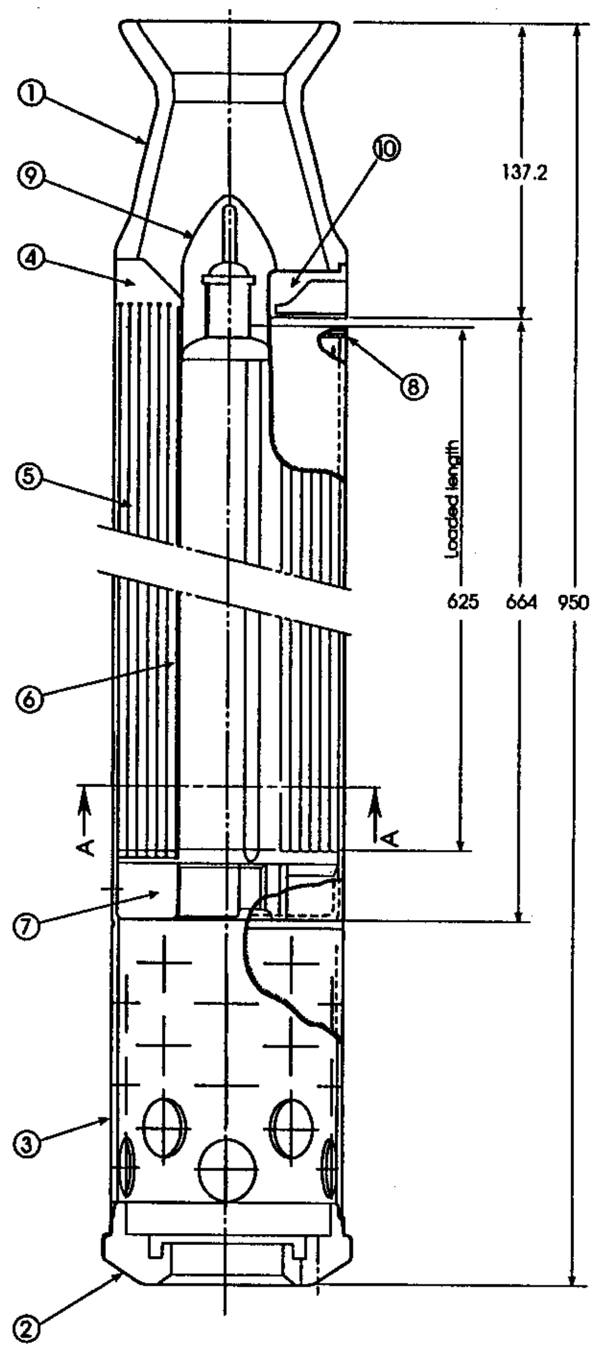


Figure B-3 Typical (Tube-Type) Aluminum-Based Fuel Element Schematic

150 g (5.3 oz) and 400 g (14.6 oz) per element. The average burnup of discharged spent nuclear fuels from these reactors ranges between 47 and 55 percent (^{235}U atom percent). The uranium enrichment used in this fuel varies from just below 20 to 93 percent.

The following provides additional information on a typical tubular type spent nuclear fuel element (shown in Figure B-3) that was used in a 10 MW reactor.

This fuel element initially contains 220 g (7.7 oz) ^{235}U, and consists of 5 concentric fuel tubes. Each tube is made of three curved fuel plates. The fuel is an alloy of uranium in aluminum with a thin (0.38 mm) aluminum cladding. Five different curved fuel plate width sizes with 1.27 mm (0.05 in) thickness and 625 mm (24.6 in) height are used. The overall outside diameter of the outermost tube is 103 mm (4 in). The plate width and the ^{235}U content for each plate size are:

Plate Number	*1*	*2*	*3*	*4*	*5*
Width (mm)	57.9	66.9	75.8	84.8	93.7
^{235}U (g)	10.70	12.70	14.60	16.60	18.60

The overall dimensions of a cut element, leaving the fuel portion intact, are 103 mm (4 in) outside diameter and 664 mm (25.4 in) in length, with an overall weight of approximately 6,000 g (13.2 lbs).

B.1.3.3 Pin-Type Design

Three types of foreign research reactors use pin-type design fuel. They are: the Canadian Safe LOW POwer critical [K] Experiment (SLOWPOKE) (20 kW power); the Canadian NRU (125 MW power) and South Korean KMRR (30 MW) reactors; and the Romanian TRIGA (14 MW) reactors. Among these reactors, the SLOWPOKE fuel pins are the smallest in size and uranium content. The NRU and KMRR reactor fuels are considered special type fuel, and the Romanian reactor fuels are TRIGA or rod-type fuel. Special-type and rod-type materials are discussed below.

The SLOWPOKE reactor fuel pins have an outside diameter of 4.73 mm (0.2 in), a length of 220 mm (8.7 in), and contain 93 percent enriched uranium fuels. The ^{235}U content of each pin is 2.8 g (0.1 oz). The maximum fuel burnup of discharged spent nuclear fuels is about 2 percent (^{235}U atom percent) in 10 to 20 years of reactor operation.

The SLOWPOKE spent nuclear fuel pins are usually bundled together in 10 to 15 pins per bundle. In the past, this fuel was shipped to Savannah River Site in 50.8-mm (2-in) outside diameter, 2.9-m- (9.6-ft-) long canisters containing between 150 to 160 pins per canister.

B.1.3.4 Special-Type Design

Special-type design fuels are used in the French RHF (57 MW power), Canadian NRU (125 MW power) and NRX (24 MW power), and the South Korean KMRR (30 MW) reactors. The fuel type in the Canadian research reactors consists of clusters of about 3-m- (9.84-ft-) long uranium aluminum alloy fuel pins clad in aluminum. The initial ^{235}U content of each fuel cluster varies between 491 g (17.3 oz) and 545 g (19.2 oz). The current operating reactor (NRU) uses a fuel element that consists of a cluster of 12 long pins containing 491 g (17.3 oz) of ^{235}U per cluster. Each fuel pin has an overall length of 296 cm (116.5 in), and the fuel portion is 274.3 cm (107.9 in) long. The fuel cluster including the flow tube is cut to a length of 292.6 cm (115.2 in) before shipment. The average burnup of discharged spent nuclear fuels from an NRU reactor is about 76 percent (^{235}U atom percent). Figure B-4 shows a 12-pin cluster NRU fuel element. The fuel in the South Korean research reactor consists of two types of fuel clusters; one is

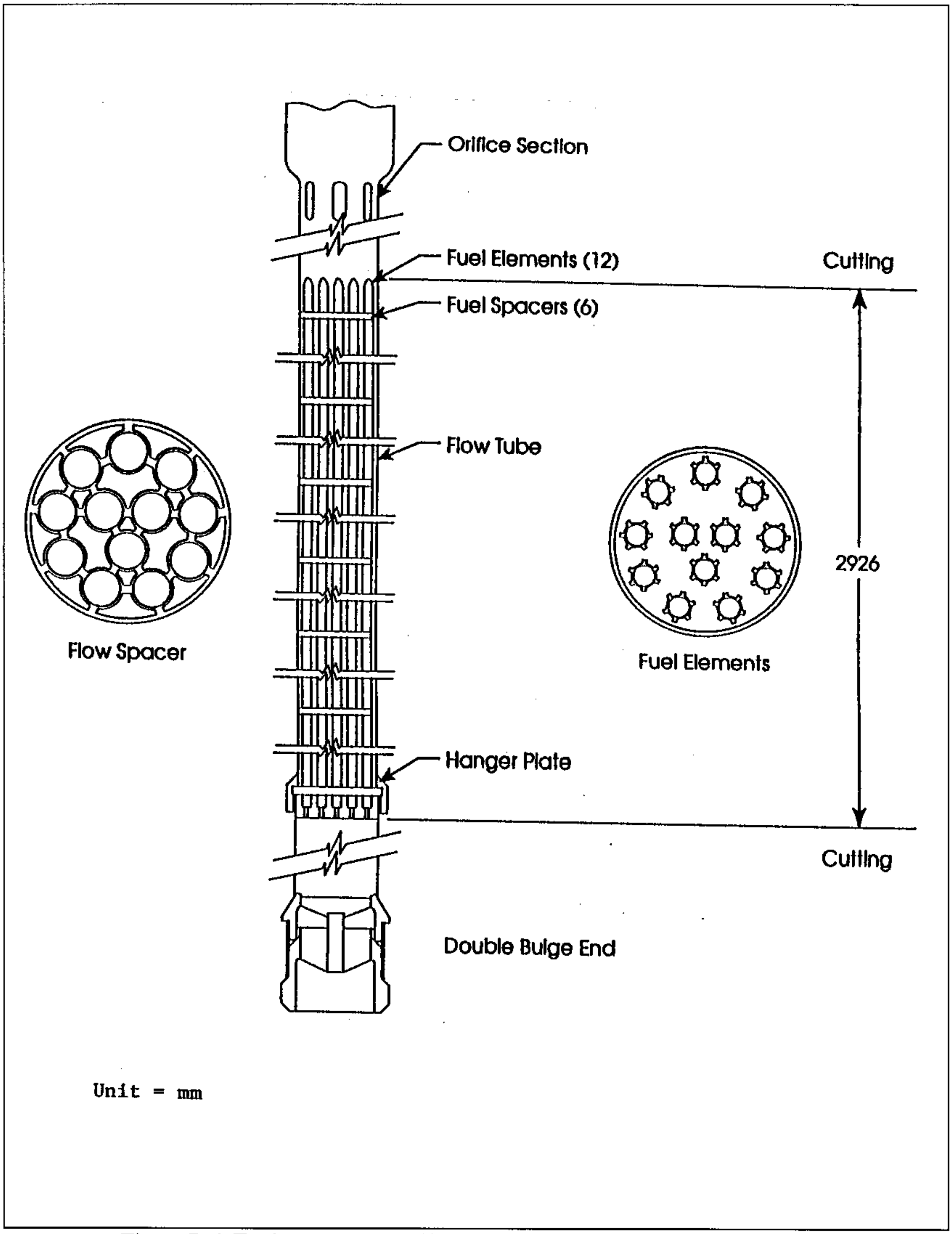


Figure B-4 Typical NRU Type (Aluminum-Based) Fuel Element Schematic

18 pins per cluster with an initial ^{235}U content of 248 g (8.7 oz). The second is 36 pins per cluster with an initial ^{235}U content of 435 g (1 lb). The expected burnup of a discharged spent nuclear fuel from this reactor is approximately 65 percent (^{235}U atom percent).

The fuel used in the RHF research reactor is an annular-type fuel element. The RHF research reactor uses only one fuel element at a time. The RHF fuel element contains 9.2 kg (20.3 lbs) of uranium, enriched to 93 percent, in ^{235}U in 280 involute fuel plates made of uranium aluminum alloy (UAl_3-Al) clad in aluminum. The weight of a cut element is about 100 kg (220 lbs). The fuel is in the annulus of two aluminum tubes: the inner tube has an outside diameter of 274 mm (10.8 in), and the outer tube has an outside diameter of 414 mm (16.3 in). The expected average burnup of a discharged spent nuclear fuel is 36 percent (^{235}U atom percent). Figure B-5 shows a schematic drawing of a configuration of annular fuel element similar to that of RHF fuels.

B.1.3.5 Rod-Type Design

This fuel type design is used in TRIGA research reactors. These research reactors have power ranging from 100 kW to 14 MW. The TRIGA fuel is mainly made up of three basic types of fuel elements: aluminum-clad elements, stainless steel-clad elements, and incoloy-clad elements. All aluminum-clad elements and stainless steel-clad elements are 38.1-mm (1.5-in) diameter by 762-mm- (30-in-) long rods including end fittings (see Figure B-6). The incoloy-clad elements are of the same length, but with a smaller diameter, ranging from 13.7 mm (0.54 in) to 30.7 mm (1.2 in). The 13.7-mm (0.54-in) fuel is currently being used in the Romanian TRIGA research reactor. The fuel is a solid, homogeneous mixture of uranium zirconium hydride alloy. A 6.35-mm (0.25-in) hole is drilled through the center of the active fuel section to facilitate hydriding; a zirconium rod is inserted in this hole after hydriding is complete.

The aluminum-clad elements are the original TRIGA fuel rods that are still in use at some foreign research reactors. The active part of the aluminum-clad fuel element contains about 8 percent by weight of uranium enriched to just below 20 percent ^{235}U. The hydrogen-to-zirconium atom ratio is approximately 1.0. The initial loading of ^{235}U is about 38 g (1.3 oz). The average burnup of this type of fuel is about 8 percent. Each rod weighs 3.2 kg (7.04 lbs) on the average.

The current standard TRIGA fuel rods are the stainless steel-clad elements. The fuel content of the stainless steel element can vary according to the type used. The fuel content of a standard rod consists of 8 to 9 percent by weight of 19.95 percent enriched uranium [about 39 g (1.4 oz) of ^{235}U] in zirconium hydride, with a hydrogen-to-zirconium atom ratio of 1.7. Another type, known as FLIP, contains 8.5 percent by weight of 70 percent enriched uranium [137 g (4.8 oz) of ^{235}U]. The annular core pulsed reactor fuel type contains 12 percent by weight of just below 20 percent enriched uranium [about 54 g (1.9 oz) of ^{235}U] in zirconium hydride with a hydrogen-to-zirconium ratio of 1.7. The expected average burnup of the discharged spent nuclear fuel is approximately 15 percent. Each rod weighs 3.6 kg (7.9 lbs) on the average.

The incoloy-clad element has a longer active fuel length [558.8 mm (22 in) compared to 381 mm (15 in) for standard stainless steel-clad]. The fuel section consists of four pellets, each 139.7-mm (5.5-in) long, and contains approximately 45 percent by weight of uranium enriched to 20 percent [approximately 54 g (1.9 oz) of ^{235}U] in zirconium hydride. There are no graphite reflectors within this element. Instead, a 76.2-mm (3-in) spring is inserted at the top and bottom of the element, and stainless steel end fixtures are attached to both ends of the can. The expected average burnup of this fuel in the Romanian TRIGA reactor is about 52 percent.

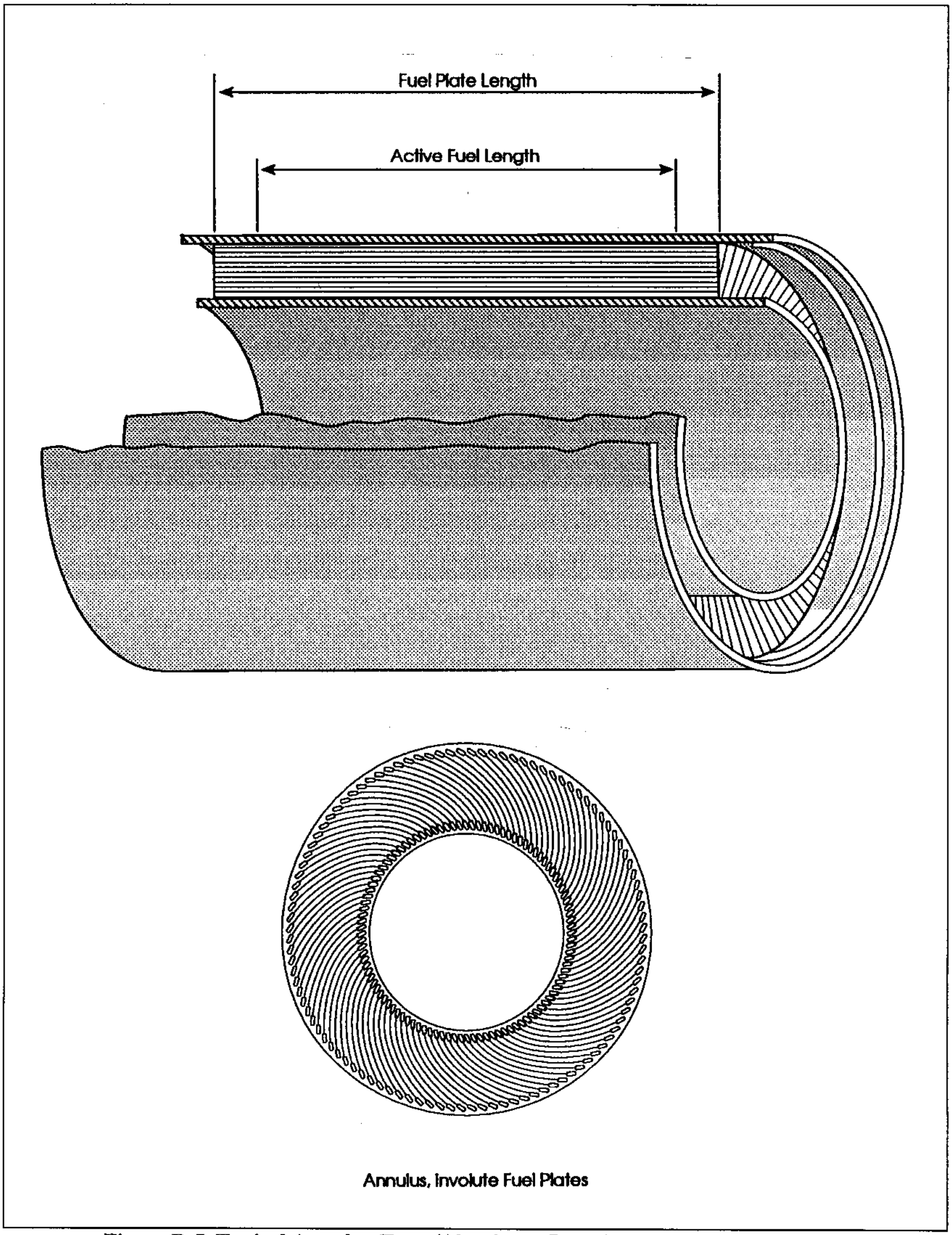
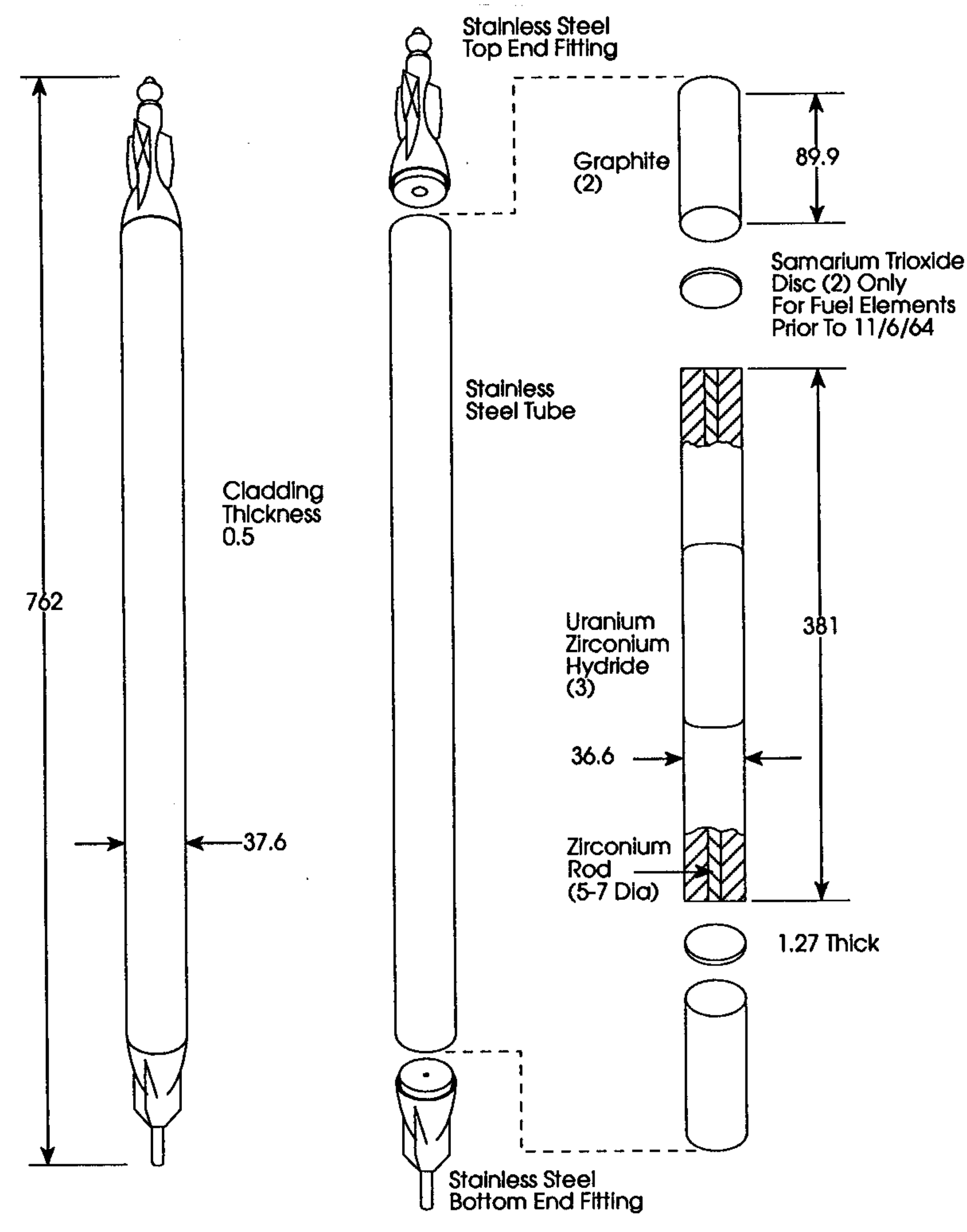


Figure B-5 Typical Annular-Type (Aluminum-Based) Fuel Element Schematic



(Fuel could have aluminum, incoloy, or stainless steel cladding)

Unit = mm

Figure B-6 Typical TRIGA Fuel Element Schematic

B.1.4 Description of the Bounding Radionuclide Inventory

The spent nuclear fuel radionuclide concentration (or inventory) is directly related to the initial mass of fuel (fissile and fertile), the level of burnup, and the cooling period (or decay period) following fuel discharge from the reactor. The fuel burnup is a function of the fuel position inside the reactor core resulting in some fuels burning more than others. A well-designed fuel management program, however, reduces burnup variations among fuel elements. The radionuclide generation in an irradiated fuel is a function of reactor power level and the duration of irradiation process. Research reactors have irregular irradiation profiles, and are typically operated at various power levels. For the calculation of the radionuclide inventory, each fuel element was assumed to have been burnt uniformly and continuously at full reactor power before its discharge. These assumptions maximize the radionuclide inventories of the spent nuclear fuel. As stated earlier, the cooling or decay time after fuel discharge from the reactor determines the amount of radionuclides that remains in the spent nuclear fuel.

Based on the discussion in Section B.1.3, the foreign research reactor spent nuclear fuels were grouped into three classes and four fuel categories for the determination of bounding radionuclide inventories. This subdivision was created in order to provide a better representation of potential radionuclide inventories associated with each type of fuel. This subdivision also provides a means for identifying the type of transportation casks needed, and estimating the number of spent nuclear fuel shipments. The radionuclide inventories per shipment are needed as input to marine and ground transportation and cask handling impact analyses.

The selected fuel types for the determination of bounding radionuclide inventories are:

1. ***Special.*** These are aluminum-based fuels that are neither TRIGA nor MTR. Special fuels are also different in size and geometry.

   1a. *Single Element Reactors*. Spent nuclear fuel from research reactors that operate with one element (e.g., RHF of France). These spent nuclear fuels contain several kg of ^{235}U and require special shipping baskets, casks, and transportation analyses.

   1b. *NRU Type Spent Nuclear Fuel.* Spent nuclear fuel from Canadian Research Laboratories' research reactors (e.g., NRU, and NRX) that require special transportation analysis. These spent nuclear fuels are geometrically different from an MTR-type and TRIGA spent nuclear fuel both in cross section and length, and require special shipping arrangements in addition to being transported overland by truck or rail.

2. ***MTR Spent Nuclear Fuel.*** This category covers all MTR-type spent nuclear fuels. These spent nuclear fuels have similar geometrical characteristics and use common type transportation casks.

3. ***TRIGA Spent Nuclear Fuel.*** Spent nuclear fuel from TRIGA reactors. These spent nuclear fuels also have almost similar geometrical characteristics and use common types of transportation casks.

In the case of special-type fuel, the bounding spent nuclear fuels are the RHF of France and the NRU of Canada. For the identification of a bounding spent nuclear fuel within the MTR and TRIGA fuel types, a series of ORIGEN2 (Croff, 1980) computer runs was made using different spent nuclear fuels within each fuel type. ORIGEN2 generates the radionuclide inventory in a spent nuclear fuel based on the fuel burnup, initial fissile and fertile inventory, and decay time. The radionuclide inventories of selected bounding spent nuclear fuel within each fuel type were determined assuming that the spent nuclear fuel has been cooled for a specified period after its discharge from the reactor. In order to maximize the radionuclide inventory per transportation cask, a review of the potential casks was performed. It was determined that the use of IU-04 (Pegase) transportation casks, maximizes the radioactive inventory and requires the shortest cooling period (maximum of 1 year) (see Section B.1.6). Based on this review, the cooling period for each bounding spent nuclear fuel was determined. The bounding spent nuclear fuels for MTR and TRIGA type fuel are found to be BR-2 type spent nuclear fuel, and a spent nuclear fuel from a 3-MW TRIGA reactor burning 31g (1.1 oz) of ^{235}U in 3 years, respectively. The bounding TRIGA spent nuclear fuel identified here is a theoretical bounding fuel for this category.

Table B-6 provides a list of the radioactive isotopes and their inventories for selected bounding spent nuclear fuel types. The list of isotopes is generated from ORIGEN2 output based on the following criteria:

1. All isotopes (from a list of 270 elements) that could have a potential to contribute 1 mrem from inhalation and ingestion are considered. The estimates of dose associated with each isotope intake were based on the effective committed dose equivalent factors provided in DOE/EH-0071 (DOE, 1988).

2. Once all isotopes were selected, those that contribute to 99.9 percent of total health hazard were chosen.

3. Isotopes such as ^{85}Kr, ^{235}U, and ^{238}U were added to the list as historically significant isotopes, although they do not meet the above criteria.

It is important to note that the radionuclide inventories identified here are for calculational purposes only. The majority of the spent nuclear fuels would have lower radionuclide inventories than what is identified here, and the likelihood of a full cask containing maximum inventory during the acceptance policy period would be low. By the time the policy would become effective in late 1995, there could be about 10,000 spent nuclear fuel elements, of which 80 percent would have had more than 2 years of cooldown (decay). The number of spent nuclear fuels that receive maximum burnup used in the estimation of the radionuclide inventory is very small when compared to the total number of the spent nuclear fuel elements estimated in each fuel category.

B.1.5 Characteristics and Radionuclide Inventories of Target Materials

Under Implementation Alternative 1 to Management Alternative 1 of the proposed action, DOE would plan to manage target material. The total amount of target material is estimated to be about 0.56 MTHM having a volume of 6.5 m^3 (230 ft^3). Target materials are residual materials from target fuels that have been irradiated in a research reactor to produce ^{99}Mo, which decays to ^{99}Tc, a medical isotope. Four countries (Canada, Belgium, Argentina, and Indonesia) use target fuel containing U.S.-origin enriched uranium for the production of medical isotopes. Canada, Argentina, and Belgium currently use aluminum-based targets containing HEU, and Indonesia currently uses a target that consists of a layer of HEU oxide (UO_2) material plated on the interior surface of a stainless steel tube. The distribution of target materials from these countries includes: 0.525 MTHM from Canada; 0.029 MTHM from Belgium; 0.0014 MTHM from Indonesia; and 0.0011 MTHM from Argentina. A target fuel is irradiated to a burnup level of about 3 percent (^{235}U atoms percent) before being discharged from the reactor. Once the target fuel is removed from the reactor, within a short period the fuel is dissolved and ^{99}Mo is separated from the solution. The residual material is then decayed. Prior to shipment, the residual materials are transformed to an acceptable form.

Table B-6 Bounding Radionuclide Inventories per Element for Selected Fuel Categories (Curies)

Isotope	*Fuel Category*			
	BR-2	*RHF*	*NRU*	*TRIGA*
Tritium	2.40	37	3.95	0.328
Krypton 85	68.7	1,070	113	9.10
Strontium 89	1,133	17,600	405	68.8
Strontium 90	578	8,930	967	79
Yttrium 90	578	8,930	967	79
Yttrium 91	2030	31,400	842	115
Zirconium 95	2,972	46,300	1,410	163
Niobium 95	6,111	94,900	3,060	320
Ruthenium 103	247	3,770	60.0	21.1
Rhodium 103m	247	3,770	60.0	21.1
Ruthenium 106	597	9,160	767	63.5
Rhodium 106m	597	9,160	767	63.5
Tin 123	11.9	184	10.0	0.978
Antimony 125	24.7	381	38.0	2.98
Tellurium 125m	5.89	90.6	9.21	0.718
Tellurium 127m	24.6	382	18.4	1.40
Tellurium 129m	5.25	79.8	0.958	0.578
Cesium 134	456	4,000	1,480	29.0
Cesium-137	572	8,870	958	79.8
Cerium 141	159	2,440	277	175
Cerium 144	8,667	135,000	10,600	633
Praseodymium 144	8,667	135,000	10,600	633
Promethium 147	1,342	24,600	1,240	175
Promethium 148m	2.10	29.2	0.0583	1.17
Europium 154	17.2	163	56.3	1.05
Europium 155	3.61	45.6	10.2	0.565
Uranium 234	0.0000254	0.000374	0.0000654	0.00000453
Uranium 235	0.000383	0.0109	0.000253	0.000199
Uranium 238	0.00000947	0.000206	0.00000111	0.000163
Plutonium 238	1.78	10.3	11.3	0.0760
Plutonium 239	0.0511	0.0889	0.0138	0.0138
Plutonium 240	0.0333	0.421	0.0101	0.0523
Plutonium 241	7.89	67.7	2.95	5.33
Americium 241	0.0110	0.0967	0.00517	0.0102
Americium 242m	0.0000292	0.000155	0.0000250	0.000225
Americium 243	0.000120	0.00376	0.000146	0.0000110
Curium 242	0.0486	0.127	0.0429	0.131
Curium 244	0.0369	0.00926	0.0113	0.000178
Total (Curies)	35,129	546,000	34,700	2,740
Thermal (Watts)	147	2,250	150	10.4

There are currently two methods for preparing the residual materials containing aluminum for transport. The first method is calcining and canning the material with the existing aluminum, and the second is a method that first removes aluminum from the residual materials and then oxidizes the remains. The final products are then canned. A process similar to the latter is used for the Indonesian target materials. Since the Indonesian target materials do not contain aluminum, no aluminum separation is needed. In this case, a precipitation process is used to separate the target materials from the solution. The precipitated materials are then dried and canned in preparation for transport.

The canned material from the first process contains 40 grams (1.4 oz) of ^{235}U per can. The second process allows a higher amount of ^{235}U, 200 g (7 oz), to be packed in a similar can. Can material could be aluminum or stainless steel. In the past, the target material was shipped to the Savannah River Site in aluminum cans 64 mm (2.5 in) in diameter and 280 mm (11 in) long. The use of the first process would result in a total 140 shipments of this material to the United States, and the second process would result in a total of 57 shipments. These number of shipments were estimated based on an assumption that the target material cans would be in transportation casks that would not contain other types of spent nuclear fuel. However, in all likelihood, with small amounts of target materials (such as Indonesia and Argentina), would not ship a partially filled transportation cask when other spent nuclear fuel could be added to fill the cask. Therefore, these estimates represent an upper bound on the total number of the target material shipments. The radionuclide inventory of a target material can containing from 40 to 200 g (1.4 to 7 oz) of ^{235}U, and that of a transportation cask containing this material, is given in Table B-7. This inventory is estimated based on 1 year decay time of the target material solution before the canning process.

B.1.6 Foreign Research Reactor Spent Nuclear Fuel Shipment Estimates

Tables B-1 and B-2 provide the estimated number of foreign research reactor spent nuclear fuel shipments from each country. These estimates were based on a set of assumptions that maximize the potential impacts from transportation. Review of the potential transportation casks identified eight casks with various capabilities (see Section B.2.2). These casks are certified to accommodate between 1 and 126 spent nuclear fuel elements per cask based on a variety of cask cavity configurations. Each transportation cask can be certified to ship different fuel types by using various baskets in the cask cavity. For example, a transportation cask like IU-04 has been certified to accommodate several different fuel types by using various baskets in the cask cavity. On the other hand, a cask like LHRL-120 is currently certified to accommodate only one specific fuel type (Australian HIFAR fuel). Based on this review, IU-04 was identified as the bounding cask (highest curies content for the number of elements shipped per cask) for the transportation accident analyses.

In an attempt to capture various types of spent nuclear fuel, maximize the amount of radionuclides per cask, and allow for potential partial cask shipments, for the purposes of the analyses in this EIS, the following assumptions were made to estimate the number of shipments for each type of fuel:

1. The number of shipments for MTR-type spent nuclear fuel elements was estimated based on 30 elements per cask. The radionuclide inventory per cask was estimated based on a full cask, that is, 36 spent nuclear fuel elements of the bounding MTR-type (BR-2 fuel) per cask. One exception: for the Australian spent nuclear fuel, cask LHRL-120 which was built specifically for this fuel was used for estimating the number of shipments. The allowed radionuclide inventory in this cask is the smallest of all casks identified. Nonetheless, each of the LHRL-120 casks was assumed to contain the same quantity of radionuclide inventories as that of a cask containing 36 elements of the bounding MTR-type spent nuclear fuel.

2. The number of shipments for NRU-type spent nuclear fuel was estimated based on 24 NRU elements per cask. The radionuclide inventories per cask were also based on 24 NRU elements per cask.

Table B-7 Radionuclide Inventories of Target Material per Can and per Transportation Cask (Curies)

Isotope	Curies for 40 g ^{235}U per Can	Curies for 200 g ^{235}U per Can	Cask Curies with 40 g per Can	Cask Curies with 200 g per Can
Strontium 89	4.06E+00	2.03E+01	1.95E+02	4.87E+02
Strontium 90	3.28E+00	1.64E+01	1.58E+02	3.94E+02
Yttrium 90	3.28E+00	1.64E+01	1.58E+02	3.94E+02
Yttrium	9.18E+02	3.84E+01	3.69E+02	9.22E+02
Zirconium 95	1.18E+01	5.90E+01	5.67E+02	1.42E+03
Niobium 95	2.53E+01	1.27E+02	1.21E+03	3.04E+03
Ruthenium 103	7.4E-01	3.72E+00	3.57E+01	8.93E+01
Rhodium 103m	7.5E-01	3.73E+00	3.58E+01	8.95E+01
Ruthenium 106	3.11E+00	1.55E+01	1.49E+02	3.73E+02
Rhodium 106m	3.11E+00	1.55E+01	1.49E+02	3.73E+02
Tin 123	6.0E-02	2.8E-01	2.70E+00	6.74E+00
Antimony 125	1.4E-01	6.8E-01	6.51E+00	1.63E+01
Tellurium 125m	3.0E-02	1.6E-01	1.56E+00	3.91E+00
Tellurium 127m	1.1E-01	5.6E-01	5.39E+00	1.35E+01
Tellurium 129m	1.0E-02	7.0E-02	6.7E-01	1.68E+00
Cesium 134	1.0E-02	6.0E-02	6.1E-01	1.53E+00
Cesium-137	3.26E+00	1.628E+01	1.56E+02	3.91E+02
Cerium 141	4.2E-01	2.11E+00	2.03E+01	5.07E+01
Cerium 144	4.53E+01	2.27E+02	2.18E+03	5.44E+03
Praseodymium 144	4.57E+01	2.29E+02	2.20E+03	5.49E+03
Promethium 147	1.07E+01	5.36E+01	5.14E+02	1.29E+03
Promethium 148m	5.06E-04	2.53E-03	2.43E-02	6.07E-02
Europium 154	1.65E-03	8.23E-03	7.90E-02	1.97E-01
Europium 155	6.97E-02	3.49E-01	3.35E+00	8.37E+00
Uranium 234	1.42E-07	7.09E-07	6.81E-06	1.70E-05
Uranium 235	8.29E-05	4.15E-04	3.98E-03	9.95E-03
Uranium 238	1.50E-06	7.52E-06	7.22E-05	1.80E-04
Plutonium 238	3.33E-06	1.67E-05	1.60E-04	4.00E-04
Plutonium 239	6.15E-04	3.08E-03	2.95E-02	7.38E-02
Plutonium 240	1.43E-05	7.13E-05	6.85E-04	1.71E-03
Plutonium 241	1.48E-04	7.38E-04	7.09E-03	1.77E-02
Americium 241	2.42E-07	1.21E-06	1.16E-05	2.91E-05
Americium 242m	4.43E-12	2.22E-11	2.13E-10	5.32E-10
Americium 243	3.07E-12	1.54E-11	1.47E-10	3.69E-10
Curium 242	1.43E-09	7.15E-09	6.86E-08	1.72E-07
Curium 244	3.40E-12	1.70E-11	1.63E-10	4.08E-10
Total (Curies)	1.69E+02	8.30E+02	7.97E+03	1.99E+04
Thermal (Watts)	6.8E-01	3.40E+00	3.26E+01	8.160E+01

3. The number of shipments for RHF type spent nuclear fuel was estimated based on one element per cask. The bounding cask can only accommodate one bounding spent nuclear fuel element per cask.

4. The number of shipments for TRIGA spent nuclear fuel was estimated based on 30 elements per cask. The radionuclide inventories per cask were based on 40 elements of bounding TRIGA spent nuclear fuel element per cask.

Table B-8 provides a list of radionuclide inventories per transportation cask for selected fuel categories.

Table B-8 Bounding Radionuclide Inventories per Transportation Cask for Selected Fuel Categories (Curies)

Isotope	*Fuel Category*			
	BR-2	*RHF*	*TRIGA*	*NRU*
Tritium	86.4	37.0	13.1	94.8
Krypton 85	2,470	1,070	364	2,710
Strontium 89	40,800	17,600	2,750	9,720
Strontium 90	20,800	8,930	3,160	23,200
Yttrium 90	20,800	8,930	3,160	23,200
Yttrium 91	73,000	31,400	4,580	20,200
Zirconium 95	107,000	46,300	6,500	33,800
Niobium 95	220,000	94,900	12,800	73,400
Ruthenium 103	8,900	3,770	844	1,440
Rhodium 103m	8,900	3,770	844	1,440
Ruthenium 106	21,500	9,160	2,540	18,400
Rhodium 106m	21,500	9,160	2,540	18,400
Tin 123	427	184	39.1	240
Antimony 125	890	381	119	912
Tellurium 125m	212	90.6	28.7	221
Tellurium 127m	887	382	55.8	442
Tellurium 129m	189	79.8	23.1	23.0
Cesium 134	16,400	4,000	1,160	35,400
Cesium 137	20,600	8,870	3,190	23,000
Cerium 141	5,740	2,440	7,000	6,650
Cerium 144	312,000	135,000	25,300	254,000
Praseodymium 144	312,000	135,000	25,300	254,000
Promethium 147	48,300	24,600	7,000	29,800
Promethium 148m	75.6	29.2	46.8	1.40
Europium 154	620	163	41.8	1,350
Europium 155	130	45.6	22.6	245
Uranium 234	0.000914	0.000374	0.000181	0.00157
Uranium 235	0.0138	0.0109	0.00794	0.00606
Uranium 238	0.000341	0.000206	0.00650	0.0000267
Plutonium 238	64.2	10.3	3.04	270
Plutonium 239	1.84	0.0889	0.551	0.332
Plutonium 240	1.20	0.421	2.09	0.242
Plutonium 241	284	67.7	213	70.9
Americium 241	0.396	0.0967	0.407	0.124
Americium 242m	0.00105	0.000155	0.00900	0.000600
Americium 243	0.00433	0.00376	0.000438	0.00351
Curium 242	1.75	0.127	5.25	1.03
Curium 244	1.33	0.00926	0.00713	0.270
Total (Curies)	1,260,000	546,000	110,000	833,000
Thermal (Watts)	5,290	2,250	416	3,600
Number of casks by January 2006	473	86	162	116

B.1.7 Amount of Foreign Research Reactor Spent Nuclear Fuel In Implementation Alternative 2a of Management Alternative 1

Under this implementation alternative (see Section 2.2.2.2), DOE would adopt an alternative policy duration of 5 years (1995-2000). The amount of spent nuclear fuel expected under this alternative is approximately 18,800 elements, containing approximately 13 MTHM, and having a volume of 87 m^3 (3,300 ft^3). Tables B-9 and B-10 provide an estimate of the total amount of spent nuclear fuel that would be available (i.e., currently stored or to be generated) in each country by January 2001 (Matos, 1994). These tables also provide the estimated number of shipments expected from each country. The breakdown of the number of shipments in terms of the four bounding fuel categories (as defined in Section B.1.4) are: 377 of BR-2, 56 of RHF, 154 TRIGA, and 91 of NRU type fuel shipments.

B.1.8 Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography

This section summarizes the estimated amount of foreign research reactor spent nuclear fuel, in terms of fuel type and geography,[4] that could be received under different implementation alternatives of Management Alternative 1 to the proposed action. The estimated amount of spent nuclear fuel for the two policy durations (i.e., a 10-year and a 5-year spent nuclear fuel generation period) are provided in Tables B-1 and B-9, for aluminum-based fuels, and in Tables B-2 and B-10 for TRIGA fuels. These tables provide a breakdown of the estimated amount of spent nuclear fuel to be accepted from each country. Table B-11 summarizes the same information given in the above tables by fuel type and geography. The information provided in this table is the basis for the calculations of transportation (ground and marine) impacts under the basic implementation of Management Alternative 1, the proposed action to manage foreign research reactor spent nuclear fuel in the United States.

DOE is also considering a Management Alternative 3, which is a hybrid of Management Alternatives 1 and 2. Under this Management Alternative as described in Section 2.4, some of the foreign research reactor spent nuclear fuels would be reprocessed overseas, and the remaining spent nuclear fuels would be brought back to be managed in the United States. Overseas reprocessing is considered only for countries that currently have the technology and capability to store research reactor fission product high- or intermediate-level wastes. The countries that can accept research reactor fission product wastes, based on the historical evidence, are: Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom. Under this Management Alternative, DOE would encourage the reprocessing of aluminum-based spent nuclear fuels from the research reactors in the above countries at western European reprocessing facilities (i.e., at Dounreay Scotland, and/or other locations) and that the recovered ^{235}U be blended down and used as LEU fuel. Reprocessing spent nuclear fuels from the above countries overseas would reduce the amount of foreign research reactor spent nuclear fuels that would be managed in the United States. Table B-12 provides a distribution of the remaining foreign research reactor spent nuclear fuels by fuel type and geography that would be brought to the United States under this Hybrid Alternative. As indicated in this table, the reduction only affects spent nuclear fuels entering through the East Coast of the United States (compare Tables B-11 and B-12). It is important to note that the existing overseas reprocessing facilities have not separated ^{235}U from TRIGA fuels. This does not mean that these facilities will not be able to process TRIGA fuels in the near future. At least one facility has stated that it has a specialty plant that can reprocess small quantities of TRIGA spent nuclear fuels (UKAEA, 1994). If this

4 *Geography refers to that amount of spent nuclear fuel that is expected to arrive at an East Coast or a West Coast port of entry to the United States. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and Eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Table B-9 Estimated Number of Aluminum-Based Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2001

Country	*Estimated Number of Spent Nuclear Fuel Elements*	*Initial Mass of Uranium, (kg)*[b]	*Estimated Number of Shipments*
Argentina[a]	283	71	9
Australia	795	247	7
Austria	130	147	4
Belgium	1,391	569	46
Brazil[a]	155	99	5
Canada	2,243	3,058	92
Chile[a]	58	12	2
Colombia[a]	16	2	1
Denmark	485	372	16
France	1,432	2,110	102
Germany	1,111	471	37
Greece[a]	199	73	6
Indonesia[a]	138	164	4
Iran[a]	29	6	1
Israel	153	34	5
Italy	150	43	5
Jamaica[a]	2	1	1
Japan	2,401	2,219	80
Korea (South)[a]	98	187	4
Netherlands	1,141	678	38
Pakistan[a]	82	16	3
Peru[a]	29	39	1
Philippines[a]	50	24	2
Portugal[a]	79	51	3
South Africa	50	10	2
Spain [c] (from Scotland)	40	16	1
Sweden	864	915	29
Switzerland	159	128	5
Taiwan	127	66	4
Thailand[a]	31	5	1
Turkey[a]	50	51	2
United Kingdom	12	4	1
Uruguay[a]	19	18	1
Venezuela[a]	120	82	4
Total	**14,122**	**11,988**	**524**

[a] *Countries other than high-income economies (World Bank, 1994). These are considered to be "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2*

[c] *40 spent nuclear fuel elements of Spain's JEN-1 reactor core are stored in Dounreay, Scotland.*

capability is acquired, then the amount of spent nuclear fuel to be managed in the United States would be lower than that indicated in Table B-12 by 834 TRIGA spent nuclear fuel elements containing 157 kg of LEU heavy metal resulting in 28 less shipments to the eastern coast of the United States by January 2005.

Table B-10 Estimated Number of TRIGA Reactor Spent Nuclear Fuel Elements of U.S.-Origin Generated by Foreign Research Reactor Operators by January 2001

Country	Estimated Number of Spent Nuclear Fuel Elements	Initial Mass of Uranium (kg)[b]	Estimated Number of Shipments
Austria	102	19	3
Bangladesh[a]	100	49	3
Brazil[a]	75	14	3
Finland	171	33	6
Germany	338	64	11
Indonesia[a]	233	44	7
Italy	343	64	11
Japan	321	61	11
Korea (South)[a]	320	61	11
Malaysia[a]	89	44	3
Mexico[a]	175	33	6
Philippines[a]	120	74	4
Romania [a]	1,451	189	48
Slovenia[a]	318	60	10
Taiwan	134	80	4
Thailand[a]	136	35	4
Turkey[a]	69	13	2
United Kingdom	89	17	3
Zaire[a]	132	25	4
Total	**4,716**	**979**	**154**

[a] *Countries other than high-income economies (World Bank, 1994). These are identified as "developing" countries.*

[b] *To derive uranium mass in pounds, multiply the amount by 2.2.*

If additional countries were to be able to accept research reactor fission product waste, additional spent nuclear fuels could be reprocessed overseas. This would reduce the amount of spent nuclear fuel to be managed in United States even further.

B.2 Transportation Casks

Spent nuclear fuel elements are transported in stainless steel packages called transportation casks, or just casks.

B.2.1 Transportation Cask Regulations

This section discusses the international and domestic regulations on transportation cask design, performance, certification, use, and transport.

B.2.1.1 International Regulations

To ensure public safety worldwide, the international community has adopted regulations for the transport of radioactive materials. The international authority for these regulations is the International Atomic Energy Agency. The emphasis of the International Atomic Energy Agency regulations for radioactive materials transport is package integrity. As promulgated in International Atomic Energy Agency Safety

Table B-11 Summary of the Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography

	January 2001					January 2006				
	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U
All Countries:										
1. Aluminum-Based	14,122	524	11,988	3,992	7,995	17,803	675	18,184	4,531	13,650
East[a]	10,395	419	9,024			13,186	544	13,919		
West[a]	3,727	105	2,963			4,617	131	4,263		
2. TRIGA	4,716	154	980	79	901	4,940	162	1,033	83	950
East	3,088	101	499			3,245	107	528		
West	1,628	53	481			1,695	55	505		
Developing Countries:										
1. Aluminum-Based	1,488	52	911	155	756	1,686	59	1,195	157	1,038
East	1,084	38	480			1,152	40	561		
West	404	14	431			534	19	634		
2. TRIGA	3,218	105	642	77	565	3,359	109	674	81	593
East	2,045	67	302			2,134	70	319		
West	1,173	38	340			1,225	39	355		

[a] *East refers to the eastern United States ports of entry. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Table B-12 Distribution of Foreign Research Reactor Spent Nuclear Fuel by Fuel Type and Geography for the Hybrid Alternative

	January 2001					January 2006				
	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U	No. of Elements	No. of Shipments	Initial Kg U	HEU Kg U	LEU Kg U
All Countries:										
1. Aluminum-Based	9,839	328	8,650	2,259	6,391	12,210	406	12,912	2,263	10,646
East[a]	6,112	223	5,687			7,593	275	8,645		
West[a]	3,727	105	2,963			4,617	131	4,263		
2. TRIGA	4,716	154	980	79	901	4,940	162	1,033	83	950
East	3,088	101	499			3,245	107	528		
West	1,628	53	481			1,695	55	505		
Developing Countries:										
1. Aluminum-Based	1,488	52	911	155	756	1,686	59	1,195	157	1,038
East	1,084	38	480			1,152	40	561		
West	404	14	431			534	19	634		
2. TRIGA	3,218	105	642	77	565	3,359	109	674	81	593
East	2,045	67	302			2,134	70	319		
West	1,173	38	340			1,225	39	355		

[a] *East Refers to the eastern United States ports of entry. Spent nuclear fuel shipments from foreign research reactors located in Europe, Africa, Middle East, and eastern part of Central and South America are designated as East Coast shipments. All others are designated as West Coast shipments.*

Series 6, radioactive materials must be transported in specially designed transportation casks that minimize the potential consequences of transportation accidents. Transportation cask designs must demonstrate their capability to ensure containment and to provide shielding by testing or analysis to the extent required by these regulations. Under International Atomic Energy Agency regulations, spent nuclear fuel transportation cask integrity must be demonstrated by successful performance during a sequence of tests that simulate accident conditions. These tests include being dropped onto an unyielding surface, dropped onto a steel post, subjected to extremely high temperatures of 800°C (1475°F) for 30 minutes, and submersed in water. Cask designs that meet these performance criteria are issued a "Certificate of Compliance" by a delegated national authority, referred to as the "Competent Authority." The Competent Authority is responsible for certifying casks that are designed or used within its "national boundary." The Competent Authority for the United States is the Department of Transportation.

To be used outside the country of origin, transportation casks must have a Certificate of Competent Authority from the country of intended use. As the Competent Authority, the Department of Transportation is responsible for granting a Certificate of Competent Authority to foreign-designed transportation casks intended for use in the United States.

B.2.1.2 Domestic Regulations

Regulations for the transport of radioactive materials in the United States are issued by the Department of Transportation, and are codified in Title 49 of the Code of Federal Regulations Parts 171-178 (49 CFR §171-178). These regulations reference accepted standards promulgated by organizations such as the International Atomic Energy Agency, the International Civil Aviation Organization, the International Air Transport Agency, the International Maritime Organization, and the U.S. Nuclear Regulatory Commission (NRC). Federal standards are updated periodically to reflect new information and to remain current with international standards, to minimize delays in international traffic, and avoid duplication of effort.

The regulation authority for radioactive materials transport is jointly shared by the Department of Transportation and NRC. As outlined in a 1979 Memorandum of Understanding with NRC, the Department of Transportation specifically regulates the carriers of spent nuclear fuel and the conditions of transport, such as routing, handling and storage, and vehicle and driver requirements. The Department of Transportation also regulates the labeling, classification, and marking of all spent nuclear fuel packages. NRC regulates the packaging and transport of spent nuclear fuel for its licensees, which include commercial shippers of spent nuclear fuel. In addition, NRC sets the standards for packages containing fissile materials and spent nuclear fuel. A detailed discussion of Federal design and performance regulations for transportation cask begins with Section B.2.1.3.

DOE policy requires compliance with applicable Federal regulations regarding domestic shipments of spent nuclear fuel. Accordingly, DOE has adopted the requirements of 10 CFR §71, "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions," and 49 CFR §171-179, "Hazardous Material Regulations." Foreign research reactor spent nuclear fuel shipments are subject to regulations set by the Department of Transportation and NRC.

B.2.1.3 Cask Design Regulations

Spent nuclear fuel is transported in robust "Type B" transportation casks that are certified for transporting radioactive materials. These transportation casks are subject to stringent design, fabrication and operating requirements imposed by the Competent Authority for the country of origin. Casks designed and certified for spent nuclear fuel transportation within the United States must meet the applicable requirements of

NRC for design, fabrication, operation, and maintenance as contained in 10 CFR §71. These regulations generally conform to International Atomic Energy Agency regulations that are presented in the International Atomic Energy Agency Safety Series 6 manual.

Cask design and fabrication can only be done by approved vendors with established quality assurance programs (10 CFR §71.101). Cask and component suppliers or vendors are required to obtain and maintain documents that prove the materials, processes, tests, instrumentation, measurements, final dimensions, and cask operating characteristics meet the design basis established in the Safety Analysis Report for Packaging for the cask, and that the cask will function as designed.

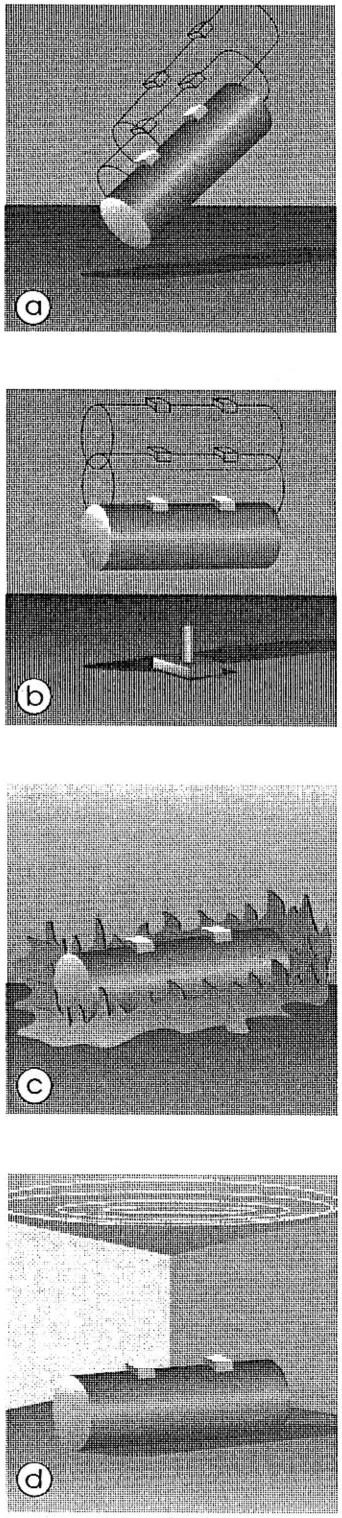
Regardless of where a transportation cask is designed, fabricated, or certified for use, it must meet certain minimum performance requirements (10 CFR §71.71-71.77). The primary function of a spent nuclear fuel transportation cask is to provide containment, criticality control, and shielding. Regulations require that casks must be operated, inspected, and maintained to high standards, ensuring their ability to contain their contents in the event of a transportation accident (10 CFR §71.87). There are no documented cases of a release of radioactive materials from spent nuclear fuel shipments even though thousands of shipments have been made by road, rail, and water transport modes. Further, a number of obsolete casks have been tested under severe accident conditions to demonstrate their adherence to design criteria without failure. Such tests have demonstrated that transportation casks are not only fabricated to a very high factor of safety; they are even sturdier than required.

Transportation casks are built out of heavy, durable structural materials, such as stainless steel. These materials must ensure cask performance under a wide range of temperatures (10 CFR §71.43). In addition to the structural materials, shielding is provided to limit radiation levels at the surface and at prescribed distances from the surface of transportation casks (10 CFR §71.47). Shielding typically consists of dense material such as lead or depleted uranium. In some cases, additional materials are added to provide neutron shielding such as water-filled outer jackets, or highly hydrogenous materials such as polyethylene. The cask cavity is configured to hold various contents including spent nuclear fuel assemblies. The assemblies are supported by internal structures or baskets that provide shock and vibration resistance, establish minimum spacing and criticality control through the use of nuclear poison materials such as boron-impregnated metals, and heat transfer to maintain the temperature of the contents within the limits specified in the Safety Analysis Report for Packaging.

Finally, to limit impact forces and minimize damage to the structural components of a cask in the event of a transportation accident, impact-absorbing structures may be attached to the exterior of the cask. These are usually composed of balsa wood, foam, or aluminum honeycomb that is designed to readily deform upon impact to absorb impact energy. All of these components are designed to work together in order to satisfy the regulatory requirements for a cask to operate under normal conditions of transportation and maintain its integrity in an accident.

Design Certification

For certification, transportation cask must be shown by analysis and/or test to withstand a series of hypothetical accident conditions. These conditions have been internationally accepted as simulating damage to transportation casks that could occur in most reasonably foreseeable accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. These accident conditions are described in Figure B-7. The NRC recently issued revised regulations, 10 CFR Part 71, governing the transportation of radioactive materials. These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations



Standards for Spent Fuel Casks

For certification by the NRC, a cask must be shown by test or analysis to withstand a series of accident conditions. These conditions have been internationally accepted as simulating damage to spent fuel casks that could occur in most severe credible accidents. The impact, fire, and water-immersion tests are considered in sequence to determine their cumulative effects on one package. A separate cask is subjected to a deep water-immersion test. The details of the tests are as follows:

Impact

Free Drop (a) – The cask drops 30 feet onto a flat, horizontal, unyielding surface so that it strikes at its weakest point.

Puncture (b) – The cask drops 40 inches onto a 6-inch-diameter steel bar at least 8 inches long: the bar strikes the cask at its most vulnerable spot.

Fire (c)

After the impact tests, the cask is totally engulfed in a 1475°F thermal environment for 30 minutes.

Water Immersion (d)

The cask is completely submerged under at least 3 feet of water for 8 hours. A separate cask is completely immersed under 50 feet of water for 8 hours.

Figure B-7 Standards for Transportation Casks (NRC, 1987)

affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than 10^6 curies be designed and constructed so that its undamaged containment system would withstand an external water pressure of 290 psi, or immersion in 200 m (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask. Except for the addition of the deep water immersion test, the regulations applicable to the transportation of foreign research reactor spent nuclear fuel are unchanged.

Under the Federal certification program, a "Type B" packaging design must be supported by a Safety Analysis Report for Packaging, which demonstrates that the design meets Federal packaging standards. The Safety Analysis Report for Packaging must include a description of the proposed packaging in sufficient detail to identify the packaging accurately and provide the basis for evaluating its design. The Safety Analysis Report for Packaging must provide the evaluation of the structural design, materials properties, containment boundary, shielding capabilities, and criticality control, and present the operating procedures, acceptance testing, and maintenance program. Upon completion of a satisfactory review of the Safety Analysis Report for Packaging by NRC to verify compliance to the regulations, a Certificate of Compliance is issued.

B.2.1.4 Transportation Regulations

To assure that the transportation cask is properly prepared for transportation, trained technicians perform numerous inspections and tests (10 CFR §71.87). These tests are designed to ensure that the cask components are properly assembled and meet leak-tightness, thermal, radiation, and contamination limits. The tests and inspections are clearly identified in the Safety Analysis Report for Packaging and/or the Certificate of Compliance for each cask. Casks can only be operated by registered users who conduct operations in accordance with documented and approved quality assurance programs meeting the requirements of the regulatory authorities. Records must be maintained that document proper cask operations in accordance with the quality requirements of 10 CFR §71.91. Reports of defects or accidental mishandling must be submitted to NRC.

B.2.1.4.1 Communications

Proper communication assists in assuring safe preparation and handling of transportation casks. Communication is provided by labels, markings, placarding, and shipping papers or other documents. Labels (49 CFR §172.403) applied to the cask document the contents and the amount of radiation emanating from the cask exterior (transport index). The transport index lists the ionizing radiation level (in mrem/hr) at a distance of 1 m (3.3 ft) from the cask surface.

In addition to the label requirements, markings (49 CFR Subpart D and §173.471) should be placed on the exterior of the cask to show the proper shipping name and the consignor and consignee in case the cask is separated from its original shipping documents (40 CFR §172.203). Transportation casks are required to be permanently marked with the designation "Type B," the owner's (or fabricator's) name and address, the Certificate of Compliance number, and the gross weight (10 CFR §71.83).

Placards (49 CFR §172.500) are applied to the transport vehicle or freight container holding the transportation cask. The placards indicate the radioactive nature of the contents. In the United States, spent nuclear fuel is a Highway Route Controlled Quantity which must be placarded according to 49 CFR §172.507. Placards provide the first responders to a traffic or transportation accident with initial information about the nature of the contents.

Shipping papers should have entries identifying the following: the name of the shipper, emergency response telephone number, description of spent nuclear fuel, and the shipper's certificate as described in 49 CFR §172 Subpart C.

In addition, drivers of motor vehicles transporting spent nuclear fuel must have training in accordance with the requirements of 49 CFR §172.700. The training requirements include: familiarization with the regulations, emergency response information, and the spent nuclear fuel communication programs required by the Occupational Safety and Health Administration. Drivers are also required to have training on the procedures necessary for safe operation of the vehicle used to transport the spent nuclear fuel.

B.2.1.4.2 Marine Transport

Relevant regulations applying to transport of spent nuclear fuel by vessel are found in 10 CFR §71 and 73, and 49 CFR §176. The U.S. Coast Guard, part of the Department of Transportation, inspects vessels for compliance with applicable regulations and requires 24-hour prenotification (33 CFR §160.207, 211, and 213).

49 CFR §171.12 (d) states that: "Radioactive materials being imported into or exported from the United States, or passing through the United States in the course of being shipped between places outside the United States, may be offered and accepted for shipment in accordance with International Atomic Energy Agency *Regulations for the Safe Transport of Radioactive Materials, Safety Series No. 6, 1985 Edition.*" Compliance with certain specified conditions of this section is required. For example, highway route controlled quantities of radioactive material must be shipped in accordance with appropriate provisions of the hazardous materials regulations and a Certificate of Competent Authority must be obtained, with any necessary revalidations. A Certificate of Competent Authority fulfills the International Atomic Energy Agency requirement for multilateral approval for a shipment of "Type B" packages in international commerce.

49 CFR §176.5 details the application of the regulations to vessels: "...this subchapter applies to each domestic or foreign vessel when in the navigable waters of the U.S., regardless of its character, tonnage, size, or service, and whether self-propelled or not, whether arriving or departing, underway, moored, anchored, aground, or while in drydock."

49 CFR §176.15 provides for enforcement of 49 CFR Subchapter C: "(a) An enforcement officer of the U.S. Coast Guard may at any time and at any place, within the jurisdiction of the U.S., board any vessel for the purpose of enforcement of this subchapter and inspect any shipment of hazardous materials as defined in this subchapter." Provision is also made in this section to detain a vessel which is in violation of the hazardous materials regulations.

The U.S. Coast Guard may accept a certificate of loading issued by the National Cargo Bureau, Inc., as evidence that the cargo is stowed in conformity with law and regulatory requirements. The National Cargo Bureau, Inc., is a nonprofit organization directed by Government and industry representatives (49 CFR §176.18). 49 CFR §176.18 authorizes inspectors of the National Cargo Bureau, Inc., to assist the Coast Guard in administering the hazardous materials regulations. Their functions are as follows:

"(1) Inspection of vessels for suitability for loading hazardous materials;

(2) Examination of stowage of hazardous materials;

(3) Making recommendations for stowage requirements of hazardous materials cargo; and,

(4) Issuance of certificates of loading setting forth that the stowage of hazardous materials is in accordance with the requirements of 46 U.S.C. 170 and its subchapter."

Detailed requirements for radioactive materials are located in 49 CFR §176 Subpart M of the Hazardous Materials Regulations. General radioactive material stowage requirements state that "(b) A package of radioactive materials which in still air has a surface temperature more than 5°C (9°F) above the ambient air may not be overstowed with any other cargo. If the package is stowed under the deck, the hold or compartment in which it is stowed must be ventilated," (49 CFR § 176.700).

Except for exclusive-use shipments, requirements relating to transport indexes state that:

". . . the number of freight containers with packages of radioactive materials contained therein must be limited so that the total sum of the transport indexes in the containers in any hold or defined deck area does not exceed 200, and:

(1) The sum of transport indexes for any individual freight container, or group of freight containers, does not exceed 50; and,

(2) Each freight container or group of freight containers is (are) handled and stowed in such a manner that groups are separated from each other by a distance of at least six m (20 ft)," [49 CFR § 176.704(c)].

Section 176.76(a) includes provision for freight containers with hazardous materials to be carried onboard a vessel in accordance with the following:

" (1) The material must be in proper condition for transportation according to the requirements of this subchapter;

(2) All packages in the transport vehicle or container must be secured to prevent movement in any direction. However, vertical restraint is not required if the shape of the packages and the stuffing pattern precludes shifting of the load;

(3) Bulkheads made of dunnage which extend to the level of the cargo must be provided unless the packages are stowed flush with the sides or ends;

(4) Dunnage must be secured to the floor when the cargo consists of dense materials or heavy packages."

Each freight container must be placarded as required by 49 CFR §172 Subpart F of the Hazardous Materials Regulations [176.76(f)].

Section 176.80 requires that radioactive materials be segregated from other hazardous materials so that they do not interact dangerously in an accident, or alternatively, requires that the radioactive material be in separate holds when stored under deck. In 49 CFR §176.83(b), a table is provided (Table II) that specifies the minimum separation distances for different classes of hazardous materials onboard a vessel. A minimum horizontal separation distance of 3 m (10 ft) projected vertically from the reference package is required. For specified hazardous materials, the "separate from" requirement means that the materials must be placed in separate holds when stowed under deck.

B.2.1.4.3 Ground Transport

Overland shipments (by railcar or by truck) are regulated by a variety of the Department of Transportation and NRC regulations dealing with packaging, notification, escorts, and communications. In addition, there are specific regulations for carriage by rail and carriage by truck.

When provisions are made to secure a package so that its position within the transport vehicle remains fixed during transport, with no loading or unloading between the beginning and end of transport, a package shipped overland in exclusive-use closed transport vehicles may not exceed the following radiation levels as provided in 49 CFR §173.441(b):

- 1,000 mrem/hr on the external package surface;
- 200 mrem/hr at any point on the outer surface of the vehicle;
- 10 mrem/hr at any point 2 m from the vertical planes projected from the outer edges of the vehicle;
- 2 mrem/hr in any normally occupied position in the vehicle, except that this provision does not apply to private motor carriers when the personnel are operating under a radiation protection program and wear radiation-exposure monitoring devices.

The shipper of record must comply with the requirements of 10 CFR §71.5 and §73.37. Section 71.5 provides that all overland shipments must be in compliance with the Department of Transportation and NRC regulations; these regulations provide for security of irradiated reactor fuel. General requirements include: providing notification to NRC in advance of each shipment, developing a shipping plan, providing escort instructions, establishing a communications center to be staffed 24 hours a day, making arrangements with local law enforcement agencies along the route for their response (if not using law enforcement personnel as escort), ensuring that the escorts are trained in accordance with Section 73.37 Appendix D, and ensuring that escorts make notification calls every 2 hours to the communications center. Additional requirements include having two armed escorts within heavily populated areas (when not in heavily populated areas, only one escort is needed) and the capability of communicating with the communications center and local law enforcement agencies through a radiotelephone or other NRC approved means of two-way voice communication.

The shipper of record, as required by 49 CFR §173.22, provides physical security measures for spent nuclear fuel shipments equivalent to those of NRC. The shipper, or the shipper's agent, provides notification for unclassified spent fuel shipments to State officials.

B.2.1.4.3.1 Rail Transport

Rail transportation requirements for radioactive materials are contained in 49 CFR §174. Briefly, for rail shipments of spent nuclear fuel the following additional requirements apply:

- railcars carrying radioactive materials must be segregated from other cars within a train, and cannot be next to other placarded hazardous materials (49 CFR §174.85) or occupied engines or cabooses; and
- hazardous materials shipments (including radioactive) must be expedited (49 CFR §174.14).

In addition, Association of American Railroad Interchange rules require that spent nuclear fuel be shipped only on railcars meeting certain construction and packaging retention requirements (AAR Rule 88A 1d). Rail routing has not been regulated by the Department of Transportation because the railroads are privately-owned companies. However, rail routes used for spent nuclear fuel shipping must be approved by NRC under a physical security plan (10 CFR §73.37).

B.2.1.4.3.2 Truck Transport

Truck transportation requirements for radioactive materials are contained within 49 CFR §177.800. In addition to requirements for securement and segregation by total transport index (50), there are road routing requirements as well. For carriage by truck, the carrier will use interstate highways or State-designated preferred routes for movement of radioactive materials in conformity with the Department of Transportation rulemaking, Docket HM-164. These regulations, found in 49 CFR Subpart D, establish routing and driver training requirements for highway carriers of packages containing "highway-route-controlled quantities" of radioactive materials. Spent nuclear fuel shipments constitute such quantities. The Department of Transportation also issues road operating requirements for radioactive materials shipments, including parking and operating rules. Primarily, these rules require trucks to stop and undergo visual inspection by the driver every 160 km (100 mi). Domestic road routing must also be approved by NRC under a physical security plan.

Many State and local governments have established their own rules, specifying such things as prenotification requirements, time-of-day restrictions, routes, and special equipment. State and local regulations that unnecessarily burden, delay, or ban shipments of radioactive materials will be preempted under the Hazardous Materials Transportation Act. The Department of Transportation rules make routing designation by appropriate State agencies enforceable by the Federal Government according to a determination by the Department of Transportation that such route designations are likely to result in further reduction of radiological risk.

B.2.2 Potential Transportation Casks

This section provides a description of the transportation casks that could be used for marine and ground transport of foreign research reactor spent nuclear fuel. The casks were identified from a review of the "Directory of National Competent Authorities' Approval Certificates for Package Design, Special Form Material, and Shipment of Radioactive Material, 1993 Edition," and the RAMPAC (radioactive material package) database for certified radioactive materials packaging (NRC, 1993). The review included only those transportation casks with current "Type B" designations for spent nuclear fuel.

B.2.2.1 Marine Transport

Table B-13 identifies the potential transportation casks for marine transport of foreign research reactor spent nuclear fuel. Each of these casks has both a certification from the country of origin and a certificate of competent authority from the Department of Transportation, which is designated as the Competent Authority for the United States. Except for the Unifetch, each of the casks has been previously used or accepted for use by DOE.

Table B-13 Proposed Transportation Casks for Marine Transport

Transportation Cask	*Certificate*	*DOE Experience*	*Country of Origin*
LHRL-120	USA/0389/B(U)F	Yes	Australia
GNS-11	USA/0381/B(U)F	Yes	Germany
TN-1	USA/0316/B(U)F	Yes	Germany
IU-04	USA/0100/B(U)F	Yes	France
TN-7 (TN-7/2)	USA/0130/B(U)F	Yes	Germany
NAC-LWT	USA/9225/B(U)F	Yes	United States
Unifetch	GB/1113/B(M)F	No	Great Britain
Goslar	USA/0094/B(M)F	Yes	Germany

Table B-14 summarizes the essential characteristics of the marine transportation casks for foreign research reactor spent nuclear fuel, such as physical dimensions, weight, type, and quantity of spent nuclear fuel elements each cask can accommodate, cooling time before shipment, maximum activity content in a cask, and the maximum initial ^{235}U content of each element. A summary of important characteristics of these casks is also provided after Table B-14.

Table B-14 Transportation Cask Design Characteristics for Marine Transport

Shipping Cask	*Weight (MT)*	*^{235}U g/Elements*	*Fuel Type*	*Number of Elements per Cask*	*Cooling Time (Days)/Activity per Cask (kCi)*	*Cask Dimensions (mm)*
LHRL-120	21.4	150-170	MTR tubular (HIFAR)	114	2,557/80	H: 3,400 D: 2,300
GNS-11	13.6	173 323	Tubular MTR Boxed-type MTR	21-28 33	180/41 180/27	H: 1,460 D: 1,185
TN-1	18.4	NA	Boxed-type MTR	126	NA	H: 2,910 D: 950
IU-04	18.9	Varied: 150-8600	Special Tubular Boxed-type MTR TRIGA Spent Nuclear Fuel	1 36-40 40-44	about one yr/1,250	H: 2,240 D: 1,880
TN-7 (TN-7/2)	25.5 (24.5)	290 8,500	Tubular-Boxed type MTR Special	60-64 2	250-1,780/2,000 310/2,000	H: 3,155 D: 1,030
NAC-LWT	23.2	17,575 Natural* 354	Commercial PWR Commercial BWR Metallic Rod Boxed-type MTR	1 - PWR 2 - BWR 15 - Met. Rods 42	730 - PWR 730 - BWR 365 - Met. Rods 1,095	H: 5,100 D: 1,120
Unifetch	16.9	405 170	Tubular Boxed-type MTR	24 40	90/45.4 90/123	H: 2,100 D: 1,800
Goslar	10.9	320	Boxed-type MTR	13	120/960	H: 1,460 D: 1,190

**Natural = Maximum initial U^{235} is 0.711 weight percent*

NA = Not Available

PWR = Pressurized Water Reactor

BWR = Boiling Water Reactor

LHRL-120

The LHRL-120 consists of a cylindrical cask surrounded by an impact limiter supported on cradles attached to a skid that is bolted to the base of a shipping container. The cask is a right circular cylinder with two concentric walls of steel for structural strength, with the annular area between the walls. The inner shell forms the containment.

The cask is built of inner and outer shells welded to the bottom end closure plate and top bolt ring, and secured by a bolted lid with a double o-ring seal. The annular space between the shells is filled with lead and supplementary lead shielding plates are provided on the bottom end closure plate and lid. The cask has two external lifting trunnions and, except for the high strength steel bolts, lead shielding, and synthetic rubber o-ring, is constructed of stainless steel plugs.

The impact limiter consists of a steel shell filled with dense polyurethane foam arranged to provide energy absorption and thermal insulation.

During transport, the cask body is completely enclosed by an impact limiter which provides both thermal and impact protection. The impact limiter is constructed in two pieces which bolt together and surround the cask body. LHRL-120 is designed for passive cooling by means of cooling tubes that penetrate the impact limiter. Tubes in the bottom half also transfer loads to the cradles. The cask and the impact limiter are secured to the skid by two tie-down straps and restraints. The skid is bolted to the base of an open conventional shipping container and is in turn enclosed by a steel weather cover fitting inside the end walls of the container and bolted to the container base. The container has standard International Standards Organization lifting arrangements and is approved under the international convention for safe containers.

The length and diameter of the cask with the impact limiter are 3.4 m (134 in) and 2.3 m (91 in), respectively. The total mass of the cask with contents, impact limiter, skid and tie-downs is 21.36 metric tons (47,080 lb) and the gross mass of the package including lift yoke, bolt tooling, tool box, weather cover and shipping container is approximately 24 metric tons (52,800 lb).

The cask was designed by Eggers, Ridelhalgh, and Partners of Columbus, Ohio for spent nuclear fuel from the High Flux Australian Research Reactor (HIFAR).

Permitted Contents:

Irradiated spent nuclear fuel elements with a minimum decay period of 7 years

Maximum number of fuel elements per package	120[a]
Maximum fuel mass	554 kg (1,200 lb)
Maximum decay heat	290 Watts
Maximum mass of both baskets (empty)	891 kg (1,965 lb)
Maximum activity of package	80,000 Ci (3.0 x 10^{+15} Bq)
Transport Index	50

[a] *The maximum number allowed is 114.*

Two identical baskets are authorized for the LHRL-120 cask in a 1 x 2 array, (i.e., in a stacked configuration). The baskets are constructed exclusively of aluminum alloys 6061 and 6063. Each basket contains 60 cells, each providing an 11-cm- (4.3-in-) diameter by 65-cm- (25.75-in-) high cylindrical cavity for each fuel assembly. Only 57 of the cells are loaded with fuel elements; the 3 center-most positions are left unloaded. The nominal wall thickness of the cell is 6.4 mm (0.25 in). Maximum mass of both baskets tiers (empty) is 891 kg (1,965 lb).

GNS-11

The GNS-11 consists of a welded stainless steel/lead construction which is tightly closed with a primary lid. The cask body can be closed at the lid region with a protection plate. The spent nuclear fuel elements fit into a fuel basket which is inserted in the cask cavity. The cask is an upright circular cylinder with two concentric walls of steel for structural strength with the annular area between the walls filled with lead for radiation shielding. The inner steel shell forms the containment.

The containment system is formed by the cask body, the primary lid including elastomer seal rings, plugs, and boltings. During transport, hood shaped impact limiters consisting of steel plates with a soft wood filling are attached to the top and bottom of the cask. In the upper region, two trunnions are screwed to the cask body for handling. The cask has the following external dimensions:

Diameter (without impact limiters)	1,185 mm (46.7 in)
Diameter (with impact limiters)	1,355 mm (53.4 in)
Height (cask body)	1,460 mm (57.5 in)
Height (with impact limiters)	1,780 mm (70 in)

The cask body is protected during transport by top and bottom impact limiters while the cask is secured vertically on its low-boy transporter. The cask weighs about 13.6 metric tons (30,000 lb). The cask can be used to ship up to 28 tubular-type MTR elements and up to 33 box-type MTR elements with initial ^{235}U enrichment of up to 93 percent. In addition, the cask can also be used to transport other types of irradiated hardware. This cask is shown in Figure B-8.

Since the temperature on the outside of the package may exceed 50°C (122°F) and the transport index can be greater than 10, the package is to be transported as a full load or as a closed load. Therefore, a maximum of two casks could be fixed in a shipping container. The cask was designed and manufactured by the German company Gessellschaft fur Nuklear-Behalte GmbH. There are currently two GNS-11 casks available for use.

Permitted Contents:

Three different fuel baskets are authorized for use with this cask. These accommodate various types and amounts of fuel:

1. A maximum of 21 or 28 (depending on the type of fuel basket used) irradiated tubular MTR fuel elements consisting of 3 to 5 concentrically arranged fuel tubes, with the following further specifications per fuel element:

Maximum initial enrichment	80 percent
Chemical form	U-Al alloy
Maximum initial mass of ^{235}U	173.4 g (6 oz)
Maximum initial quantity of uranium	217.0 g (7.5 oz)
Maximum active length	61 cm (24 in)
Maximum diameter of outer tube	10.3 cm (4 in)
Minimum cooling time	180 days

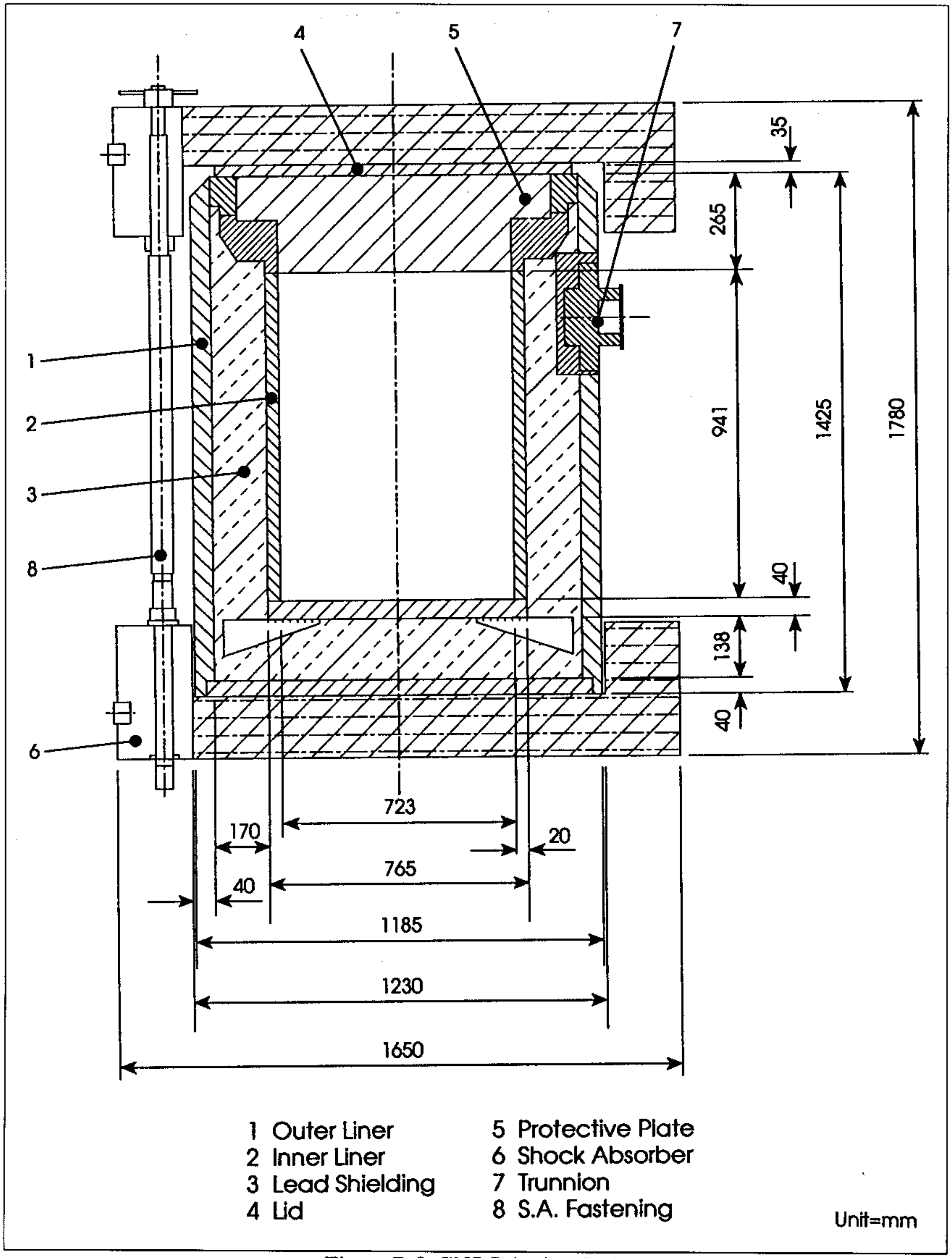


Figure B-8 GNS Shipping Cask

Thermal power (average)	maximum 76 Watts (for 21 fuel elements per cask)
Thermal power (average)	maximum 57 Watts (for 28 fuel elements per cask)
Maximum activity	40.5 kCi (1.5 PBq)

2. A maximum of 33 irradiated boxed-type MTR fuel elements, each containing a maximum of 23 aluminum-based fuel plates, with the following further specifications per fuel element:

Maximum initial enrichment	93 percent
Maximum initial mass of ^{235}U	268 g (9.3 oz)
Maximum initial mass of uranium	335 g (11.6 oz)
Active length	maximum 61 cm (24 in)
Cross-sectional area	approx. 81 x 76 mm (3.2 x 3.0 in)
Minimum cooling time	180 days
Thermal power (average)	maximum 48.5 Watts
Maximum activity	27 kCi (1 PBq)

3) A maximum of 33 irradiated boxed-type MTR-fuel elements each containing a maximum of 23 aluminum-based LEU fuel plates (containing dispersed U_3Si_2 or U_3O_8) with the following further specifications per fuel element:

Maximum initial enrichment	20 percent
Maximum initial mass of ^{235}U	323 g (11.2 oz)
Maximum initial mass of uranium	1,635 g (3.6 lbs)
Active length	maximum 61 cm (24 in)
Cross-sectional area	approx. 81 x 76 mm (3.2 x 3 in)
Minimum cooling time	360 days
Thermal power (average)	maximum 48.5 Watts
Activity	maximum 27 kCi (1 PBq)

TN-1

The TN-1 is a cylindrical double-walled steel container with lead and plaster for shielding. It is constructed of steel structural shells with the annulus between them filled with lead for gamma shielding and plaster as a heat shield. Additional heat insulation and impact resistance is provided by impact limiters. The cask, with the impact limiters, weighs 18.37 metric tons (40,500 lb). The internal cavity can accommodate three baskets, one on top of the other, each filled with up to 42 boxed-type MTR fuel elements of initial ^{235}U enrichment of up to 94 percent.

The containment system is formed by the cask body, lid with its "elastomer" seals and bolts, and three sealing plates with "elastomer" seal rings in the cask body via the quick connections.

Physical dimensions of TN-1 are as follows:

	Without Shock Absorber	*With Shock Absorber*
Width	950 mm (37.4 in)	1,284 mm (50.6 in)
Height	920 mm (36.2 in)	1,254 mm (49.4 in)
Length	2,910 mm (114.6 in)	3,075 mm (121 in)

TN-1 is designed by the French company Cogema. It is shipped in the horizontal position with the top and bottom impact limiters attached. The TN-1 cask is shown in Figure B-9.

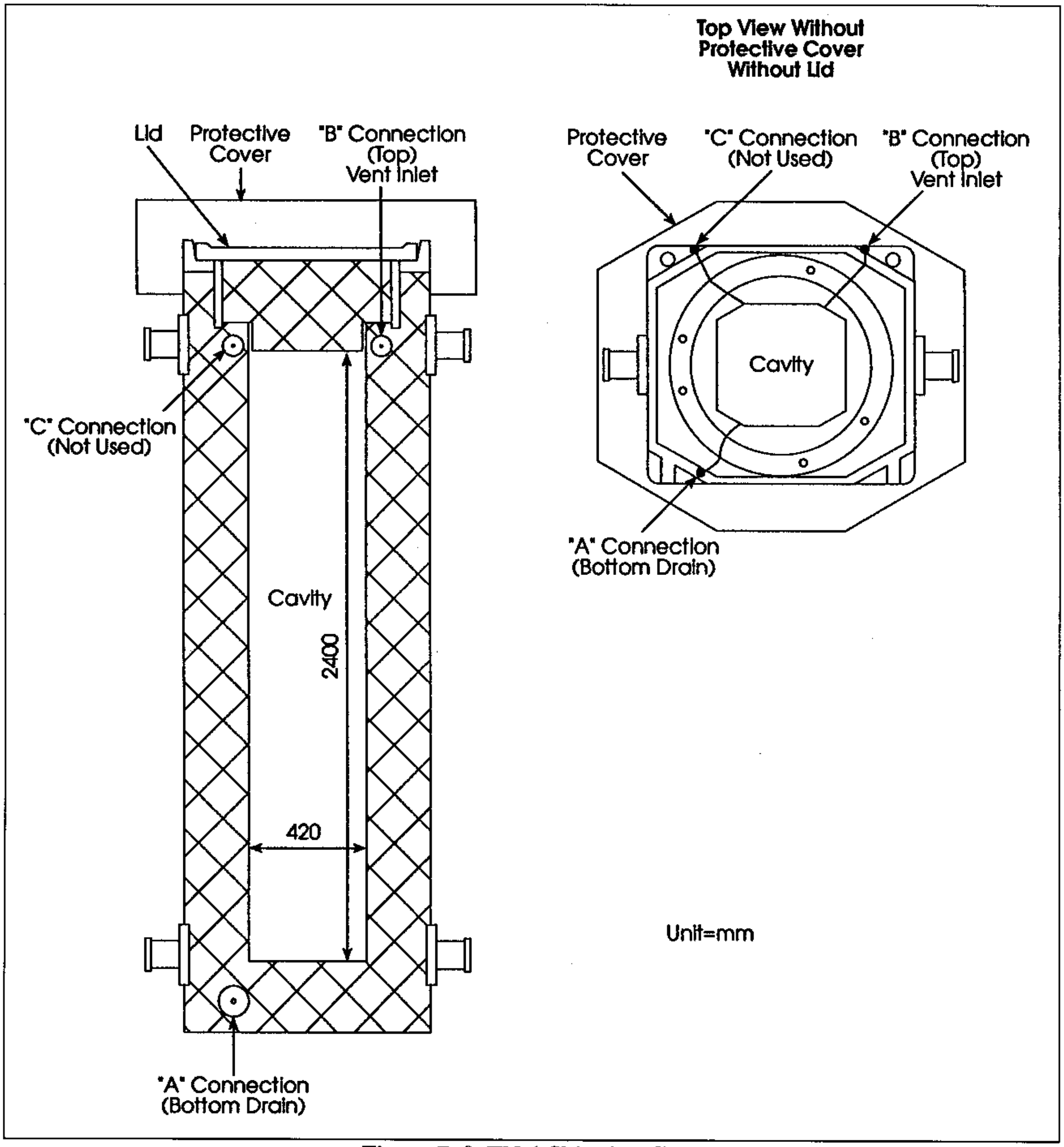


Figure B-9 TN-1 Shipping Cask

IU-04

The IU-04, also known as Pegase, consists of a body built of two stainless steel shells enclosing a lead shield. The inner confinement shell and the lead shield form a solid unit constituting the body. The outer shell is provided with a base plate filled with asbestos and is fitted with cooling fins. A layer of plaster is placed between the bottom of the outer shell and the lead. The steel lid is filled with lead and plaster. There are two pipe systems connecting the inner tank to the outside.

The cask is a right circular cylinder with two concentric walls of steel for structural strength with the annular area between the walls filled with lead for radiation shielding. The inner steel shell forms the containment. The IU-04 inner cavity can accommodate a variety of baskets which may be used to transport both MTR and TRIGA spent nuclear fuel. The cask weighs approximately 18.9 metric tons (41,670 lb). It is transported in the vertical position on the pallet, with top and bottom impact limiters to protect it in the event of an accident.

There is a main protective cover of stainless steel filled with balsa wood of two different densities. There are covers of mild steel protecting the pipe outlets which are filled with plaster. Like TN-1, this cask was also designed by Cogema. The IU-04 cask is shown in Figure B-10.

The cask is authorized to be used with various baskets designed for different types of spent nuclear fuel. The following summarizes a selected number of baskets designed for IU-04 casks:

1. Basket AA-267 - consists of cylindrical aluminum grid, 960 mm (37.8 in) high, containing 40 channels of square cross section, 84 x 84 mm (3.3 x 3.3 in), and 4 channels of cross section, 72 x 72 mm (2.83 x 2.83 in). The grid is surrounded by an aluminum belt with outside diameter of 795 mm (31.3 in). The aluminum contains two percent boron. The bottom end is covered by a 15-mm- (0.6-in-) thick aluminum plate welded to the cylindrical belt. It contains drain orifices.

Diameter	795 mm (31.3 in)
Total Height	1,030 mm (40.6 in)
Useful Height	960 mm (37.8 in)
Approx. Weight	360 kg (793.7 lbs)

   A total of 44 MTR boxed-type (72 x 72 mm cross section) fuel elements can be put in this basket. The maximum allowed residual thermal power per element is less than 80 Watts.

2. Basket TN-9083 - consists of a block of stainless steel containing five percent boron. The basket is 895 mm (35.2 in) long and has 36 lodgments of 81 x 87 mm (3.2 x 3.4 in) cross section, bored to a diameter of 98 mm (3.9 in). The bottom is covered by a plate, 12 mm (0.5 in) thick, fastened to the block by screws. The bottom plate contains drain orifices of 50 mm (2 in) diameter. The dimensions of the basket are as follows:

Base Height	90 mm (3.5 in)
Diameter	796 mm (31.3 in)
Total Height	907 mm (35.7 in)
Useful Height	895 mm (35.2 in)
Approx. Weight	1,410 kg (3,108 lbs)

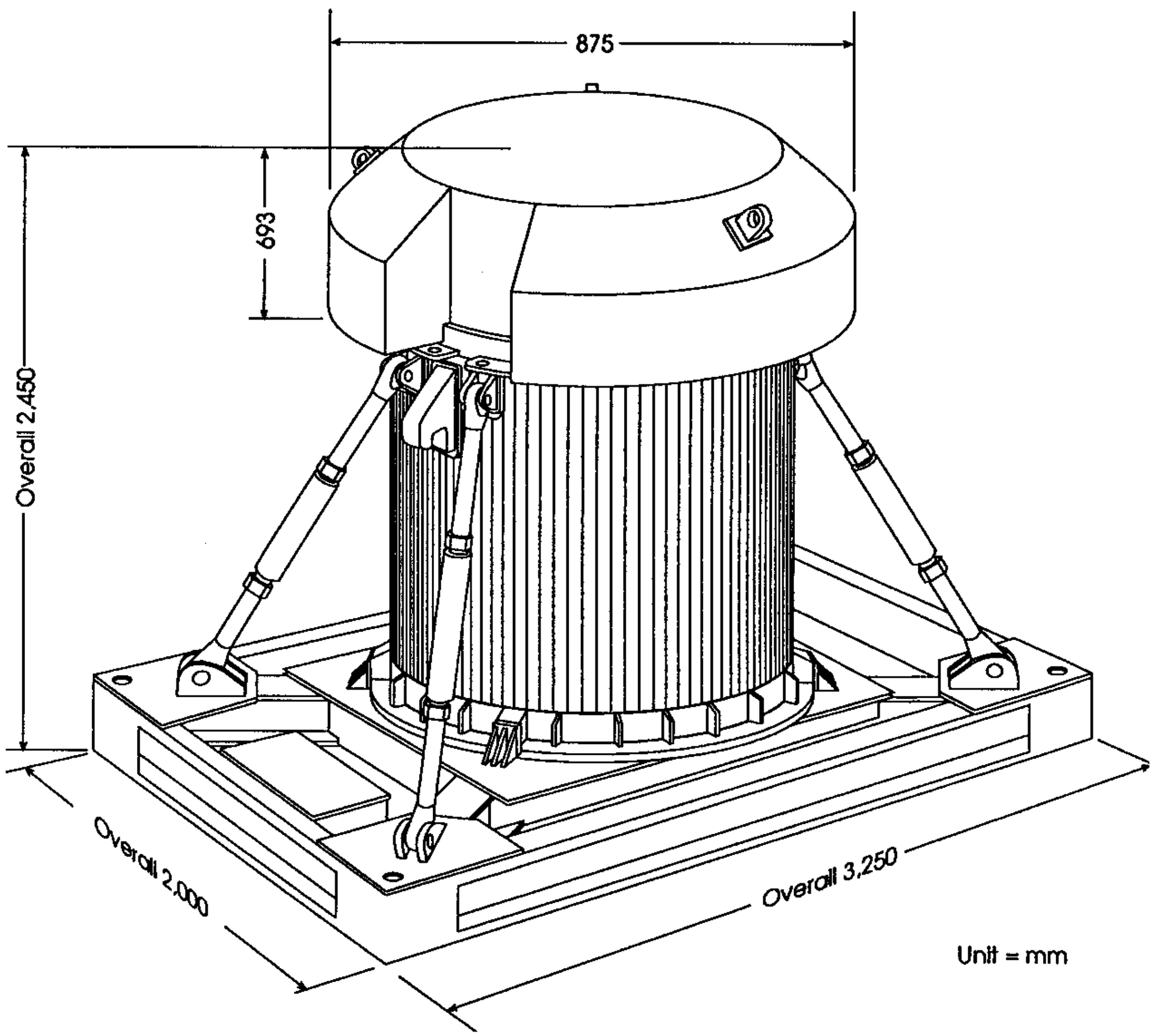


Figure B-10 IU-04 Shipping Cask

A maximum of 36 MTR tubular type fuel elements can be put in this basket. The maximum residual thermal power of each spent nuclear fuel must be less than 132 Watts.

Basket TN-9083 can also be used for TRIGA spent nuclear fuel. The maximum ^{235}U content of each TRIGA spent nuclear fuel element must be less than 40 g (1.4 oz).

3. Basket AA-49 - consists of 5 sectors of Copper-Cadmium alloy (at least 2 percent Cadmium by weight) with 5 square channels of 84 x 84 mm (3.3 x 3.3 in) and 1 channel of 71.5 x 71.5 mm (2.8 x 2.8 in), and a central core in stainless steel with a system for fastening the sectors.

Diameter	800 mm (31.5 in)
Total Height	1,030 mm (40.6 in)
Useful Height	970 mm (38.2 in)
Approx. Weight	2,500 kg (5,511 lbs)

Basket AA-49 accommodates 30 fuel elements of 93 percent enrichment from BR-2 with a maximum allowable residual power of 266 Watts per element.

4. Basket AA-50 - consists of 6 sectors of Copper-Cadmium alloy (at least 2 percent Cadmium by weight) with 6 channels of rectangular cross section 86 x 77.5 mm (3.4 x 3.1 in), and a central core in stainless steel with a system for fastening the sectors.

Diameter	800 mm (31.5 in)
Total Height	1,030 mm (40.6 in)
Useful Height	970 mm (38.2 in)
Approx. Weight	1,996 kg (4,400 lbs)

Basket AA-50 accommodates 36 boxed-type MTR fuel elements of up 93 percent enrichment. Maximum allowable residual power per each element is 200 Watts.

5. Basket AA-117 - is fabricated in Z2-CN-18-10 stainless steel with a base plate 10 mm (0.4 in) thick drilled with water drain holes in the center, 4 vertical posts 10 mm (0.4 in) thick bolted to the base plate and connected together by 3 circular spacers. Basket AA-117 accommodates 1 fuel element of 93.5 percent enrichment from RHF with a maximum allowable residual thermal power of less than 3,000 Watts.

Diameter	797 mm (31.4 in)
Total Height	1,030 mm (40.6 in)
Useful Height	420 mm (16.5 in)
Approx. Weight	165 kg (364 lbs)

TN-7 (TN-7/2)

The TN-7 consists of a cylindrical stainless steel exterior container with corresponding stainless steel lid with an integrated lead shielding; four trunnions; one bottom shock absorber; and a stainless steel, concentric cylindrical interior container, which together with its lead constitutes the "tight enclosure." Between the interior and the exterior container there is a lead shielding, 185 mm (7.3 in) thick at the sides, and 170 mm (6.7 in) thick at the lid. This shielding is surrounded by a humid cement thermal insulation. Within the interior container, up to four racks can be stacked upon each other for the admissible contents mentioned above.

The cask is a right circular cylinder with two concentric walls of steel for structural strength. The annular area between the steel walls is filled with lead for radiation shielding. The inner steel shell forms the containment.

The cask weighs about 25.5 metric tons (56,220 lb). The TN-7 was originally designed for the transportation of short light water reactor spent nuclear fuel but has the capability to accommodate the highly-enriched MTR spent nuclear fuel. In this capacity, the 4 baskets that fit in the inner cavity can accommodate up to 15 tubular or 16 box-type MTR fuel elements each.

The cask is transported in the horizontal position with top and bottom impact limiters providing protection in the event of an accident.

The TN-7/2 is very similar in design and dimension to the TN-7. The TN-7/2 is used to transport the same types and quantities of spent nuclear fuel as the TN-7. In addition, it can be used to transport up to 64 box-type MTR spent nuclear fuel elements or 2 RHF special spent nuclear fuel elements. The TN-7/2 is transported the same way as the TN-7. There is one TN-7 cask available for use at the present time. This cask has been designed by the German company Transnuklear GmbH.

Permitted contents:

1) Up to four insert racks, containing per rack:

   - maximum 15 irradiated tubular-type MTR fuel elements, each containing a maximum of 250 g (8.7 oz) of uranium enriched between 80 and 93 percent with a maximum of 200 g (6.9 oz) of ^{235}U in the form of a U-Al alloy, with a minimum cooling time of 250 days and a maximum activity of 40 kCi (1.48 PBq), or

   - maximum 16 irradiated Boxed-type MTR fuel elements, each containing a maximum of 363 g (12.6 oz) of uranium enriched between 80 and 93 percent, with a maximum of 290 g (10.1 oz) of ^{235}U in the form of a U-Al alloy, with a minimum cooling time of 1,780 days and a maximum activity of 20 kCi (740 TBq).

The racks can be combined within a cask, provided that the maximum thermal powers do not exceed 125 Watts per fuel element; 1,125 kW per rack; and 4.5 kW per cask.

OR

2) Up to two irradiated RHF type fuels, or a fuel containing a maximum number of 280 fuel plates each, with an active fuel length of about 900 mm (35.4 in), containing originally a maximum of 9.32 kg (20.6 lbs) of uranium enriched to 93 percent of ^{235}U with a maximum of 8.67 kg (19.1 lbs) of ^{235}U in the form of a U-Al alloy per element.

Maximum activity per fuel element	1,000 kCi (37 PBq)
Thermal output per fuel element	maximum 2.25 kW
Cooling time	310 days

TN-7 is authorized as Fissile Class II with a minimum Transport Index of 8.3 per package.

NAC-LWT

NAC-LWT is a steel encased lead shielded transportation cask. The cask body consists of a 19-mm- (0.75-in-) thick stainless steel inner shell, a 146-mm- (5.75-in-) thick lead gamma shield, a 30-mm- (1.2-in-) thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded to a 101.6-mm- (4-in-) thick stainless steel bottom and forging. The cask bottom consists of a 76.2-mm- (3-in-) thick, 52.7-cm- (20.75-in-) diameter lead disk enclosed by a 88.9-mm- (3.5-in-) thick stainless steel plate and bottom end forging. The cask lid is a 287-mm- (11.3-in-) thick ring stainless steel stepped design, secured to a 362-mm- (14.25-in-) thick ring forging with twelve 25.4-mm- (1-in-) diameter bolts. The cask seal is a metallic O-ring. A second teflon O-ring and a test port are provided to leak test the seal. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and teflon O-rings. The cask weighs about 22.4 metric tons (51,200 lb) including a maximum of 1.75 metric tons (4,000 lb) weight of fuel and basket.

The neutron shield tank consists of a 6.1-mm- (0.24-in-) thick stainless steel shell with 12.7-mm- (0.50-in-) thick end plates. The neutron shield region is 416.5 cm (164 in) long and 127 mm (5 in) thick. The neutron shield tank contains an ethylene glycol/water solution that is 1 percent boron by weight.

The overall dimensions of the package, with impact limiters, are 5.9 m (232 in) long by 165.1 cm (65 in) diameter. The cask cavity is 4.52 m (178 in) long and 340 mm (13.4 in) in diameter, having a volume of about 0.41 m^3 (14.5 ft^3). The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 165.7 cm (65.25 in) and a maximum thickness of 71.9 cm (28.3 in). Both impact limiters extend 30.5 cm (12 in) along the side of the cask body. The cask is transported in the horizontal position.

NAC-LWT is designed to transport one pressurized water reactor assembly, two boiling water reactor assemblies, up to 15 metallic fuel rods, or 42 boxed-type MTR foreign research reactor spent nuclear fuel with a proper basket design. There are several NAC-LWT casks available which could be used to transport foreign research reactor spent nuclear fuel. It is designed by the Nuclear Assurance Corporation in the United States.

Unifetch

Unifetch was originally designed for the transport of the spent nuclear fuel from the BR-2 (Belgium reactor). The cask weighs about 18.6 metric tons (41,000 lbs) and can accommodate either 24 or 40 spent nuclear fuel elements. The cask is transported in the vertical position. Unifetch is designed by Transport Technology in the United Kingdom.

Permitted Contents:

Two types of baskets are designed for Unifetch:

1) Baskets with maximum capacity of 24 fuel elements: Irradiated BR-2 nuclear fuel elements, assembled from plates, consisting of an inner core of natural or enriched uranium alloyed with aluminum contained within an aluminum cladding.

Fuel core thickness	0.51 mm (0.02 in)
Maximum pre-irradiation mass of ^{235}U	405 g (14.1 oz)
Maximum mass per unit length of ^{235}U/assembly	5.495 g/cm (0.5 oz/in)
Maximum decay heat per fuel element	10.7 Watts
Maximum decay heat per package	260 Watts
Minimum fuel active length	737 mm (29 in)
Maximum fuel active cross section	5,384.56 mm^2 (8.35 in^2)
Cladding thickness	0.38 mm (0.01 in)
Maximum activity of package	45.4 kCi (1.68 PBq)
Minimum cooling time	90 days

2) Baskets with maximum capacity of 40 fuel elements: Irradiated MTR boxed type nuclear fuel elements, assembled from plates, consisting of an inner core of natural or enriched uranium alloyed with aluminum contained within an aluminum cladding.

Maximum mass of ^{235}U per element	170 g (6.7 in)
Maximum mass of ^{235}U in the shield	1,265 g (2.8 lbs)
Maximum decay heat per fuel element	11.5 Watts
Maximum decay heat per package	460 Watts
Minimum fuel active length	58.42 cm (23 in)
Maximum activity of package	123.3 kCi (4.56 PBq)
Minimum cooling time	90 days

GOSLAR

The Goslar cask is a double-walled right circular cylindrical steel container that uses lead shielding in the annulus between the inner containment and outer structural container. The Goslar-Behatler was previously used to transport boxed-type MTR elements with ^{235}U enrichment between 20 percent and 93 percent from several foreign research reactors to the United States.

Goslar was designed and fabricated by Transnuklear GmbH. It weighs approximately 10.9 metric tons (24,000 lb) and has inner cavity dimensions of 483 mm (19 in) diameter x 960 mm (37.8 in) tall. Exterior dimensions, including impact limiters, are 1,185 mm (46.4 in) diameter and 1,460 mm (57.4 in) height.

Permitted Contents:

Three different fuel configurations are authorized to be used with this cask. These accommodate various types and amounts of fuel:

1) A maximum of 13 irradiated MTR fuel elements (consisting of flat or curved fuel plates) with the following further specifications per fuel element:

Maximum initial enrichment	93 percent
Chemical form	U-Al alloy
Maximum initial mass of ^{235}U	320 g (11.1 oz)
Minimum cooling time	120 days
Thermal power	maximum 300 Watts
Maximum activity	89.2 kCi (3.3 PBq)
Thermal power	maximum 3,200 Watts per cask
Maximum activity	960 kCi (35.5 PBq) per cask

2) A maximum of 13 irradiated boxed-type MTR fuel elements, with the following further specifications per fuel element:

Maximum initial enrichment	45 percent
Maximum initial mass of ^{235}U	323 g (11.2 oz)
Minimum cooling time	120 days
Thermal power	maximum 300 Watts
Maximum activity	89.2 kCi (3.3 PBq)
Thermal power	maximum 3,200 Watts per cask
Maximum activity	960 kCi (35.5 PBq) per cask

3) A maximum of 13 irradiated MTR fuel elements with a total of 10.4 kg (22.9 lbs) of uranium enriched between 17 to 80 percent with a maximum of 1.755 kg (3.9 lbs) of ^{235}U, with the following further specifications per fuel element:

Maximum initial enrichment	80 percent
Maximum initial mass of ^{235}U	135 g (4.7 oz)
Minimum cooling time	200 days
Thermal power	maximum 1 Watt
Activity	maximum 300 Ci (.0111 PBq)

B.2.2.2 Ground/Intersite Transport

Table B-15 identifies the transportation casks for ground/intersite transport of foreign research reactor spent nuclear fuel. Each of these casks has a valid certificate for use in the United States. Although some of these transportation casks are not currently certified for the shipment of research reactor spent nuclear fuel similar to that from foreign research reactors, it is anticipated that all of the casks could be recertified to accept such material.

Table B-15 Transportation Casks for Ground Transport

Transportation Cask	Certificate Number	DOE Experience	Country of Origin
NLI-10/24	USA/9023/B( )F	No	United States
IF-300	USA/9001/B( )F	Yes	United States
BMI-1	USA/5957/B(U)F	Yes	United States
GE-2000	USA/9228/B(U)F	No	United States
TN-8	USA/9015/B( )	Yes	Germany
NLI-1/2	USA/9010/B( )F	Yes	United States
NAC-LWT	USA/9225/B(U)F	Yes	United States

Design information for ground transportation casks is summarized in Table B-16. Additional narrative summary information on each of theses casks is also provided below. Although no numbers are given for each cask capacity in terms of number of foreign research reactor spent nuclear fuel elements, it has been estimated that the space for each pressurized water reactor element (assembly) can accommodate 12 to 16 foreign research reactor spent nuclear fuel elements.

NLI-10/24

The Nuclear Assurance Corporation NLI-10/24 is a railcar transported stainless steel transportation cask. The cask is 519.4 cm (204.5 in) long, 234.8 cm (96 in) diameter, and weighs 72.5 metric tons (159,000 lb) empty. Radioactive shielding is provided by lead, water, depleted uranium, and a high temperature polymer. The cask is authorized to contain either 10 pressurized water reactor or 24 BWR irradiated uranium-oxide fuel assemblies.

Table B-16 Transportation Cask Design Characteristics for Ground Transport

Transportation Cask	*Empty Weight (metric tons)*	*Fuel Type*	*Number of Elements /Cask*	*Decay Heat Generation (kW)*	*Cooling Time (Days)*	*Cask Dimensions (mm)*
NLI-10/24[a]	77.5	PWR or BWR	10 PWR/ 24 BWR	70	150	H: 5,995 D: 2,440
IF-300[a]	43.1	PWR or BWR	7 PWR/ 17 BWR	11.7	120	H: 5,335 D: 1,625
BMI-1	9.9	MTR boxed-type	24	1.5	90	H: 1,864 D: 856
GE-2000	12.7	HFIR[b] Irradiated fuel	1	0.6	120	H: 3,340 D: 1,829
TN-8[a] (TN-9)[a]	16.3 (16.3)	PWR (BWR)	3 (7)	35.5 (24.4)	150 (150)	H: 5,740 D: 1,700
NLI-1/2	21	PWR or BWR	1 PWR/ 2 BWR	10.6/ 10.6	150/ 120	H: 4,953 D: 1,200
NAC-LWT[a]	23.2	PWR or BWR MTR	1 PWR/ 2 BWR 15	2.5/ 1.1 1	730 365	H: 5,080 D: 1,120

[a] *Currently does not have proper certification for foreign research reactor spent nuclear fuel use.*

[b] *High Flux Isotope Reactor fuel is similar to that of RHF fuel.*

PWR = Pressurized Water Reactor
BWR = Boiling Water Reactor

IF-300

The General Electric IF-300 is a stainless steel encased, depleted uranium transportation cask. The cask is 533.4 cm (210 in) long, 162.6 cm (64 in) in diameter, and weighs 43.1 metric ton (95,000 lb) empty. Radioactive shielding is provided by depleted uranium, stainless steel, and a water-ethylene glycol mixture. The cask is permitted to ship 7 pressurized water reactor or 17 boiling water reactor irradiated uranium-oxide fuel assemblies. The IF-300 transportation cask is illustrated in Figure B-11.

BMI-1

The BMI-1 cask is a truck transported, steel-encased, lead shielded transportation cask. The basic body is a right circular cylinder measuring 1.86 m (73.37 in) high and 0.85 m (33.37 in) in diameter. The cask weighs about 9.9 metric tons (21,860 lb) empty. The cask is permitted to ship 24 MTR boxed-type irradiated fuel assemblies. DOE, the authorized user of the BMI-1, lends it almost exclusively for the domestic shipment of research reactor fuel. As such, its design includes eight licensed basket and canister combinations, including one for TRIGA fuel with an initial enrichment up to 93 percent. These fuels are very similar to those used by the foreign research reactors. The BMI-1 cask is illustrated in Figure B-12.

GE-2000

The GE-2000 is a truck transported, stainless steel transportation cask. It is constructed from stainless steel shells and uses lead as a shielding material. The cask is 3.34 m (131.5 in) long, 1.8 m (72 in) in diameter, and weighs about 12.7 metric tons (28,000 lb) fully loaded. Current authorized contents include irradiated fuel rods and by-product, source, or special nuclear material. The GE-2000 cask is used primarily for domestic shipments of research reactor spent nuclear fuel. It is currently being certified for

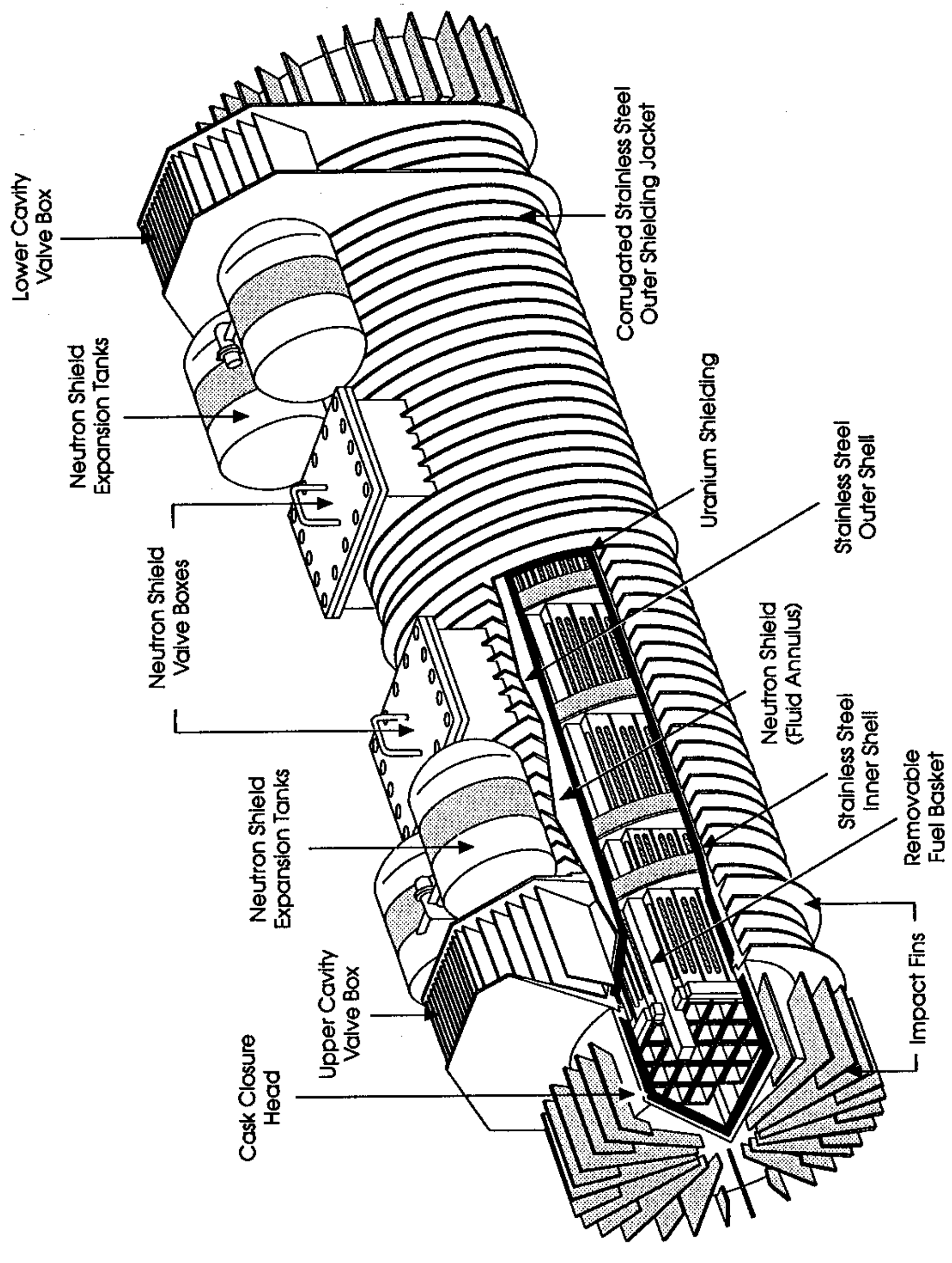


Figure B-11 IF-300 Shipping Cask

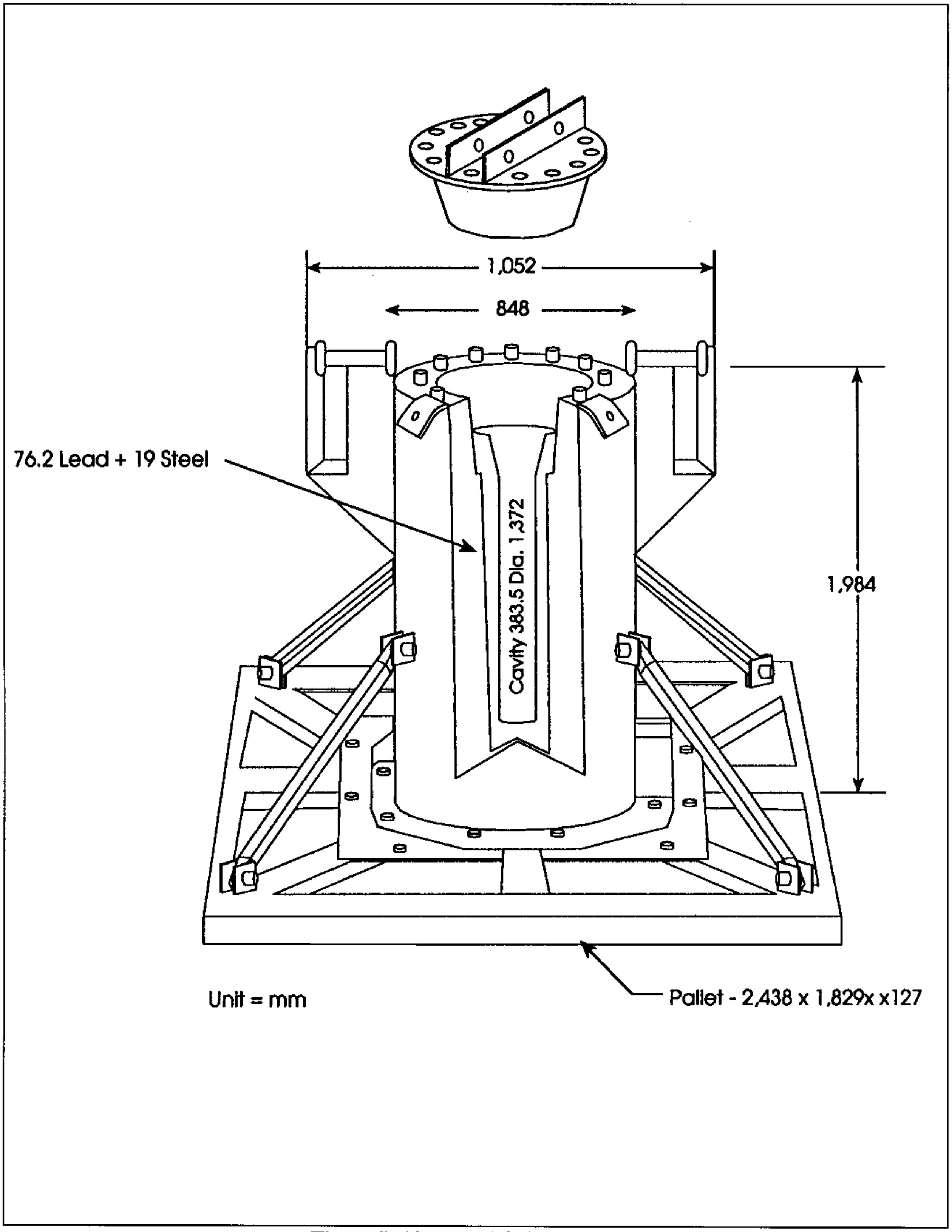


Figure B-12 BMI-1 Shipping Cask

use at the Oak Ridge National Laboratory for shipments of high flux isotope reactor fuel, which is almost similar in geometry to that used in RHF (see Section B.1.3) reactor but contains more ^{235}U fuel. The GE-2000 is illustrated in Figure B-13.

TN-8 (TN-9)

The Transnuclear TN-8 is a lead, steel, and resin shielded right cylinder, stainless steel transportation cask. The cask is 561.3 cm (221 in) long, 170 cm (67 in) in diameter, and weighs 16.3 metric tons (36,000 lb) empty. The TN-8 is permitted to ship three pressurized water reactor irradiated fuel assemblies. The TN-9 transportation cask is nearly identical to the TN-8, however, it is permitted to ship seven BWR irradiated fuel assemblies. These casks are classified as overweight truck casks in highway transport.

NLI-1/2

The Nuclear Assurance Corporation NLI-1/2 is a depleted uranium, water, and lead shielded transportation cask, encased in stainless steel. Shielding is provided by depleted uranium, lead, and a borated water-ethylene glycol mixture. The cask measures 495.3 cm (195 in) long, 120 cm (47.125 in) in diameter, and weighs 21 metric tons (49,250 lb) empty. It is permitted to ship either 1 pressurized water reactor or 2 boiling water reactor irradiated fuel assemblies. The NLI-1/2 is a legal weight truck cask that has been used at the Savannah River Site for the receipt of Taiwanese foreign research reactor spent nuclear fuel as recently as 1990. The NLI-1/2 is illustrated in Figure B-14.

NAC-LWT

The Nuclear Assurance Corporation NAC-LWT is a truck transported, steel-encased, lead shielded transportation cask. Radioactive shielding is provided by stainless steel and lead. The cask measures 508 cm (200 in) long, 165.1 cm (65 in) in diameter, and weighs 22.4 metric tons (51,200 lb) full. The cask is permitted to ship either one pressurized water reactor or two boiling water reactor irradiated fuel assemblies. This cask is also certified for the transport of Taiwanese foreign research reactor spent nuclear fuel. The NAC-LWT is nearly identical to the NLI-1/2.

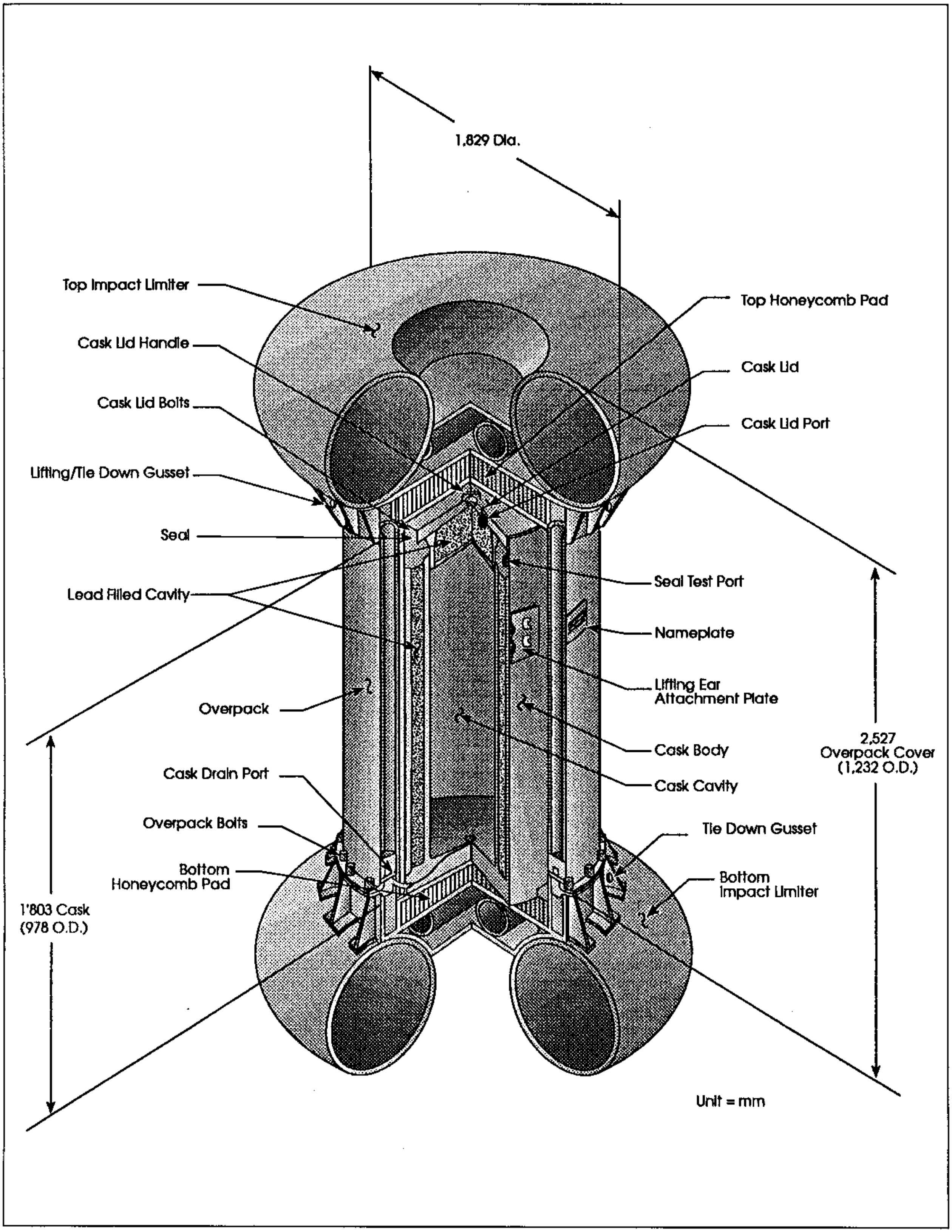


Figure B-13 GE-2000 Shipping Cask

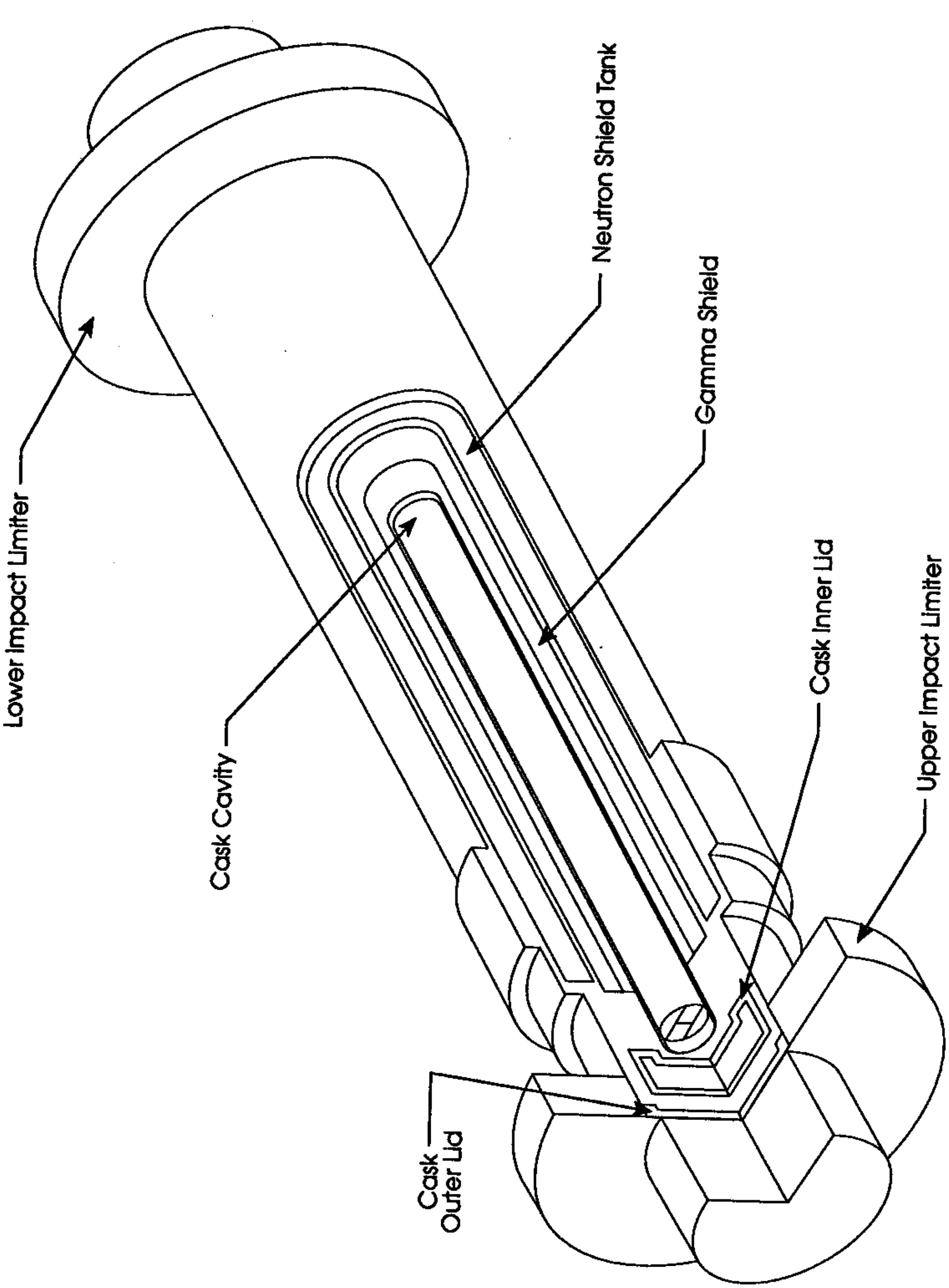


Figure B-14 NLI-1/2 Shipping Cask

References

Croff, A. G., 1980, *ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, TN, July.

DOE (U.S. Department of Energy) 1988, *Internal Dose Conversion Factors for Calculation of Dose to the Public*, DOE/EH-0071, Washington, DC, July.

DOE (U.S. Department of Energy) 1987, "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels," 52 FR 49198, Washington, DC, December 30.

DOE (U.S. Department of Energy) 1986, "Receipt and Financial Settlement Provisions for Nuclear Research Reactor Fuels," 51 FR 5754, Washington, DC, February 18.

IAEA (International Atomic Energy Agency), 1993, *Directory of National Competent Authorities' Approval Certificates for Package Design, Special Form Material and Shipment of Radioactive Material*, IAEA-TECDOC-723, 1993 Edition.

IAEA (International Atomic Energy Agency) 1989, *Directory of Nuclear Research Reactors*, Vienna, Austria.

Matos, J. E., 1994, *Foreign Research Reactor Irradiated Nuclear Fuel Inventories Containing HEU and LEU of U.S. Origin*, ANL/RERTR/TM-22, Argonne, IL, December.

NRC (U.S. Nuclear Regulatory Commission), 1995, *Nuclear Regulatory Commission, 10 CFR Part 71, Compatibility with International Atomic Energy Agency (IAEA)*, 60 FR 50248, September 28.

NRC (U.S. Nuclear Regulatory Commission), 1993, *Certificates of Compliance for Radioactive Materials Packages, Office of Nuclear Material Safety and Safeguards*, NUREG-0383, Vol. 2, Revision 16, October.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Transporting Spent Fuel Protection Provided Against Severe Highway and Railroad Accidents*, NUREG/BR-0111, Washington, DC, March.

UKAEA (United Kingdom Atomic Energy Authority), 1994, *Capability of Dounreay Fuel Plants*, Government Division, Nuclear Sites Operations, Scotland, May.

World Bank, 1994, World Development Report, *Infrastructure for Development*, Oxford University Press, New York, NY, June.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix C
Marine Transport and Associated Environmental Impacts

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Tables

List of Figures

Appendix C
Marine Transport and Associated Environmental Impacts

C.1 Introduction

Shipment of any material via ocean transport entails risks to both the ship's crew and the environment. The risks result directly from transportation-related accidents and, in the case of radioactive or other hazardous materials, also include exposure to the effects of the material itself.

This appendix provides a description of the approach used to assess the risks associated with the transport of foreign research reactor spent nuclear fuel from a foreign port to a U.S. port(s) of entry. This appendix also includes a discussion of the shipping configuration of the foreign research reactor spent nuclear fuel, the possible types of vessels that could be used to make the shipments, the risk assessment methodology (addressing both incident-free and accident risks), and the results of the analyses. Analysis of activities in the port(s) is described in Appendix D.

The incident-free and accident risk assessment results are presented in terms of the per shipment risk and total risks associated with the basic implementation of Management Alternative 1 and other implementation alternatives. In addition, annual risks from incident-free transport are developed.

C.2 Scope

This appendix addresses the modes of marine transportation and the nonradiological and radiological risks associated with marine transportation.

Transportation Modes: Marine transport of foreign research reactor spent nuclear fuel could occur via a combination of four types of vessels: container ships, roll-on/roll-off vessels, general cargo (breakbulk) vessels, or purpose-built vessels. In the incident-free analysis, it was assumed that all shipments would be made on breakbulk vessels. Breakbulk cargo vessel speeds are typical of the four types of cargo vessels considered, which means that the breakbulk vessel time enroute, (i.e., from port of origin to port of entry) is representative of the four vessel types. The ship speed selected for the analysis, 15 knots or 17.3 mph, is at the lower end of the range of speeds for commercial cargo vessels. This, in turn, maximizes the radiation dose received by the ship's crew, which bounds the incident-free risk. No vessel type assumption is necessary for the analysis of the impacts associated with the accident conditions, since these impacts are essentially independent of the type of ship.

Nonradiological Impacts: These risks were assessed as resulting in a negligible impact on the health of the public and workers. The limited number of shipments (less than a thousand individual spent nuclear fuel containers) would not result in a significant change in the number of ocean crossings by transport vessels. Regardless of the ship selection – general cargo, container, roll-on/roll-off, or purpose-built vessel – a negligible increase in the exposure of the public to exhaust emissions or transportation-related accidents would occur.

More than 56,000 port calls of ships engaged in foreign trade are made at U.S. ports each year (DOC, 1994). The basic implementation of Management Alternative 1 would result in the addition of less than 50 round trip voyages by vessel per year; the actual number of voyages that might occur would be dependent on the manner in which the policy, if adopted, was implemented. On average, less than

60 foreign research reactor spent nuclear fuel casks would be required to be shipped each year to fulfill the basic implementation shipping needs. These shipments could be made on regularly scheduled commercial cargo vessels. Alternatively, these shipments could be made in a chartered vessel, where the transportation casks would be the only cargo onboard the vessel.

If commercial cargo vessels were used, the shipment of foreign research reactor spent nuclear fuel transportation casks would not result in additional voyages specifically for the transport of the foreign research reactor spent nuclear fuel. The approximately 60 transportation casks per year would be part of the general cargo carried by the ships. As discussed in Section C.3.1.2, container vessels typically have a capacity in the range of 800 to 1,000 containers, while some carry many more. General cargo vessels tend to be somewhat smaller, but still have capacities equivalent to several hundred containers. Each foreign research reactor spent nuclear fuel transportation cask is assumed to be shipped within a container. Therefore, for the tens of thousands of vessels received at U.S. ports each year, each carrying hundreds of containers, or their equivalent, the basic implementation alternative would add approximately 60 containers per year. This is equivalent to much less than the capacity of one cargo vessel.

If chartered vessels were to be used for the shipment of the foreign research reactor spent nuclear fuel, the number of shipments required per year would depend on the number of transportation casks loaded into each vessel. Many factors would affect this number, such as the size of the ship, the availability of the ship, originating point for the shipments, and the readiness of foreign research reactor operators to ship the spent nuclear fuel. Estimates of the number of transportation casks that could be shipped on a single vessel are in the range of two to eight. This range results in estimates of between 30 and less than 10 shipments per year. Thirty shipments involve less than 0.001 of the total number of port calls by vessels engaged in foreign trade received at U.S. ports each year.

A combination of the two means of shipping the foreign research reactor spent nuclear fuel, commercial cargo and charter vessels, would result in somewhat fewer additional voyages by cargo vessels than the use of dedicated vessels alone. The use of five chartered voyages (carrying eight casks each) in combination with commercial cargo vessels could result in more than half of the foreign research reactor spent nuclear fuel casks being transported on chartered vessels. These five chartered voyages would represent less than 0.0001 of the number of vessels received at U.S. ports.

Regardless of the types of ships selected, there would be negligible impact on the marine environment including endangered species or habitats because of the negligible increase in ship traffic.

Radiological Impacts: The risks resulting from the radioactive nature of the shipments are addressed for both incident-free and accident transportation conditions. The radiological risks associated with the incident-free shipping conditions would be the potential exposure of the members of the crew to external radiation in the vicinity of the packaged fuel. No other public exposure is considered, due to the relative isolation of the material from the general public during all phases of the marine transport of the spent nuclear fuel. The potential exposure to radiation due to accidents is assessed for the marine environment in the event of the loss of a cask at sea and the consequent release of the cask's inventory into the marine environment. Only the marine exposure pathway is considered in detail, as the relative isolation from land and populated areas of the material during almost all of the voyage would minimize direct exposure through air pathways. Additionally, since the damaged cask is assumed to be lost at sea (and if not lost at sea, any airborne release would be deposited on the ocean surface), the marine pathway is likely to have more severe consequences.

All radiologically-related impacts on humans are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (EDE), which is the sum of the EDE from the external radiation exposure and the 50-year committed EDE from internal radiation exposure. The EDE is the sum of the tissue and organ-weighted dose equivalents for all irradiated tissues and organs. The committed EDE considers the initial exposure and the effects of radioactive decay and elimination of the radionuclide through ordinary metabolic processes over the 50-year period. Radiation doses are presented in units of person-rem for collective population and rem or mrem (equal to 0.001 rem) for individuals. The impacts are further expressed as health risks, primarily in terms of latent cancer fatalities (LCFs). The health risk conversion factors were derived from International Commission of Radiological Protection Publication 60 (ICRP, 1991). See Chapter 4 of this Environmental Impact Statement (EIS) for a more detailed discussion of radiation dose and risk.

C.3 Selection of Modes and Routes

C.3.1 Modes of Transportation

This section describes the possible shipping configurations of the cask and the types of vessels that could be used for ocean transport. In general, the shipping configuration of the cask conforms to the type of vessel to be used in ocean transport. The purpose of this section is to assist in understanding the specific operations or handling issues that arise in the various cask shipping configurations or in the use of specific vessel types.

Currently, the preferred method of commercial transport aboard ocean vessels is to mount casks in metal containers, sometimes called "International Standards Organization containers." Typically, containerized casks are transported on smaller general cargo vessels rather than on large vessels specifically designed for container transport.

As described in Section C.3.1.2, non-containerized transport is feasible, but is not generally used. An exception is the shipment of casks in purpose-built ships, which are specifically designed to accommodate radioactive material casks. Purpose-built ships for cask transport are described in Section C.3.1.2.

C.3.1.1 Cask Transport Configurations

This section describes the three configurations of casks for transport. The casks may be containerized, mounted on a wheeled trailer, or free-standing. Typically, containerized casks are mounted in a 6.1-m (20-ft) container, since casks rarely exceed 5.8 m (19 ft) in length. Wheeled cask trailers are usually dedicated trailers that have unique hardware used to secure the cask to the trailer frame. Free-standing casks are mounted on a skid, pallet, or cradle to facilitate handling the cask in intermodal transfer and in stowage.

Containerized Cask Configuration: Casks may be transported within International Standards Organization containers to take advantage of standardized port container lifting gear and vessel and transporter container tiedowns. The International Standards Organization container is a steel box that conforms to a set of standard dimensions, and has standard tiedown and lift points. The standard height and width is 2.4 m (8 ft). There are two standard lengths, 6.1 m and 12.2 m (20 ft and 40 ft). The four corners of the container are structural posts that have lifting points at the top and tiedown points at the bottom. These containers are commonly used to move all manner of goods transported by vessel and, because of the standardized dimensions and lifting points, can be rapidly transferred between the dock and the vessel.

Lifting, stowage, and transfer of containers is described in Appendix D.

Casks are mounted within the container using specially designed supports in the container floor. These supports mate with the tiedown structure of the cask to secure it to the container.

Figure C-1 shows a spent nuclear fuel cask being loaded into an International Standards Organization container. Containers may be either completely enclosed using a removable top, as shown in Figure C-1, or have open sides and top. Usually, an enclosed container is used with a cask that is certified for transport with a "personnel barrier." As its name implies, the personnel barrier is a structure that surrounds the cask in transport, to preclude inadvertent personnel contact with the cask surface. The barrier is a required feature if the cask surface can exceed about 52°C (125°F) in non-exclusive-use transport. The cask may become warm in transport due to the decay heat of the spent nuclear fuel within the cask. Usually, the barrier is constructed of expanded metal screen or other lightweight material. Casks that do not require a barrier may be mounted in open containers. In either case, the floor of the container is specially designed to support the weight of the cask, and to incorporate the tiedown fixtures of the cask. The tiedowns may be unique, as those shown in Figure C-1, or they may be bolts that secure the skid, pallet, or cradle to the floor of the container.

Since the introduction of International Standards Organization containers, shipment of spent nuclear fuel in casks mounted in containers has become the preferred configuration. Use of containers provides an improvement in the ease of securing the cask to the vessel. It also permits the use of standard container handling and transport equipment that is used at many ports.

Roll-On/Roll-Off Cask Configuration: Casks can be transported by vessel on a wheeled trailer that allows the cask to be rolled onto the vessel, and at the destination, rolled off. The cask (on its own unique, dedicated trailer) is moved on and off the vessel using a standard truck tractor or wheeled tug across a ramp extending between the vessel and the dock.

A few shipments have been made to the United States from Europe using casks mounted on their own dedicated trailers. However, current Federal regulations (49 CFR 176.76(b)) restrict trailered hazardous cargo (such as spent nuclear fuel) to transport on a trailership (roll-on/roll-off), trainship, ferry vessel, or car float. This regulation would preclude shipment of trailered casks containing spent nuclear fuel on general cargo, or other vessels. It has been assumed that the foreign research reactor spent nuclear fuel will be shipped as containerized cargo, not mounted on trailers. Use of containers will not limit the type of vessel that can be selected for transport.

Free-Standing Cask Configurations: Casks could be transported as a free-standing package. In this configuration, the cask would be mounted on a skid, pallet, or cradle to facilitate both lifting and tiedown. A pallet is usually required because casks have unique tiedowns and lift points that may not be readily accommodated by more common rigging and stowage bindings. The pallet is usually designed to provide a means of attaching the cask to the transport trailer or railcar. The cask is usually either attached to the pallet by bolting at the cask tiedown fixtures, or by the use of specially designed turn buckle cables.

Free-standing casks have previously been transported on general cargo vessels that carry cargo as "breakbulk." Breakbulk cargo is any cargo that is handled individually and may be containerized or otherwise unitized.

Shipments of free-standing casks are no longer routinely made, primarily because the securing of the cask to the vessel is considered to be somewhat less certain than that obtained with International Standards Organization containers, and because of the risk of damage to the cask in handling and stowage.

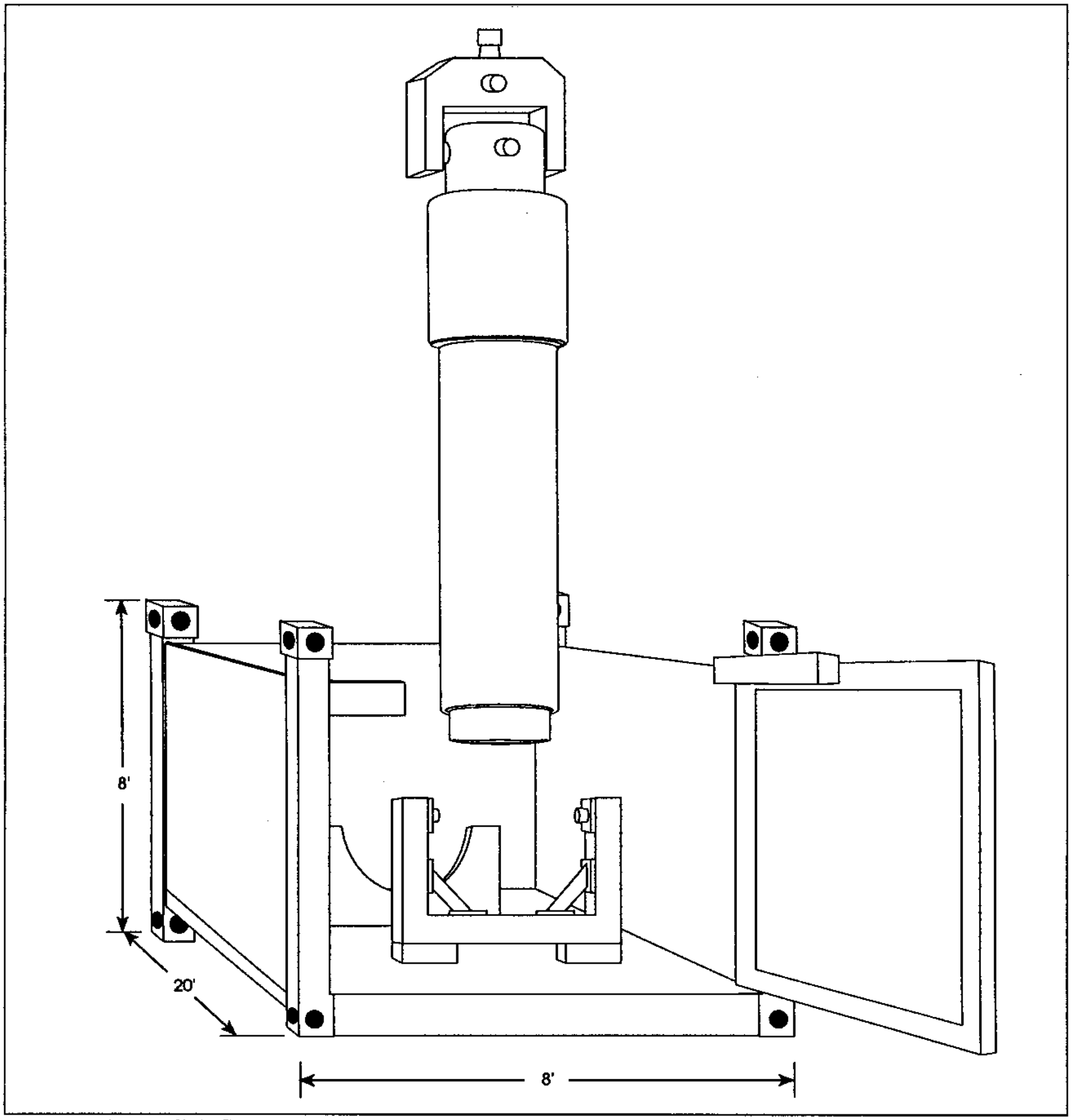


Figure C-1 Spent Nuclear Fuel Cask Being Loaded into an International Standards Organization Container

Recently, several purpose-built ships have been placed in service that transport casks in a free-standing (non-containerized) configuration. Purpose-built vessels are described in Section C.3.1.2. These dedicated vessels incorporate holds containing structural tiedowns designed to mate with the cask, and which provide additional shielding from radiation. The purpose-built vessels are operated by crews both trained in radiological safety and with a radiological control program in place.

C.3.1.2 Vessel Types, Cask Handling Requirements, and Methods of Service

This section describes the four principal types of vessels that could be used for the transport of casks. The vessel types include container, roll-on/roll-off, general cargo (also called breakbulk), and purpose-built vessels.

Each of these types of vessel have somewhat different handling requirements for the cargo they carry. Cask handling and equipment requirements are also described.

Individual shipments could be made by scheduled commercial vessel, or by charter vessel. Vessels on scheduled routes generally call on the more important ports. Scheduled vessels also typically call at intermediate ports between a given origin and destination.

Because of the general public aversion to nuclear materials, there has been a marked decrease in the number of steamship lines that will accept spent nuclear fuel cargoes in scheduled service. Also, many foreign ports and some U.S. ports do not currently permit docking or handling of spent nuclear fuel shipments, either en route or as a destination. This has led to an increased reliance on spent nuclear fuel ocean transport by chartered vessel. Vessels for charter are available from any number of steamship lines. Generally, smaller general cargo (breakbulk) vessels are used for charter shipments.

Container Vessels: Container vessels are typically large ships that are specifically intended for the transport of International Standards Organization containers (Figure C-2). Modern container ships can transport up to about 5,000 containers, although a more typical capacity is in the range of 800 to 1,000. A principal advantage of container vessels, because of standardization of containers, is that the vessel can be rapidly loaded or off loaded at those ports equipped with container gantry cranes. Containers can be removed from (or placed on) the vessel at an average rate of about 45 containers per hour. At well equipped container vessel ports, two cranes are used to move containers. Smaller container vessels may be equipped with an onboard crane allowing calls at ports that are less well equipped.

Because of cost, the only container ships generally used to transport spent nuclear fuel are in scheduled service. Smaller general cargo vessels are more suitable to chartered service, and these vessels accommodate containers.

Roll-On/Roll-Off Vessels: Roll-on/roll-off vessels are vehicle carriers (Figure C-3) used for the ocean transport of cars and trucks. The vessels are loaded and unloaded using a ramp between the vessel and dock. Ordinarily, the vessel carries its own ramp, which is deployed by an on-board crane, hydraulic cylinders, or chain drives. The ramp may extend from the stern of the vessel or from a hatch in the side hull of the vessel. At docks intended for roll-on/roll-off service, additional ramps may be deployed from the dock to expedite loading or unloading. For ocean transport, the trailers are lashed to the deck(s) of the vessel using ratchet or turnbuckle type bindings to fixed securement points in the deck. It is likely that a roll-on/roll-off capable vessel could be leased, should a roll-on/roll-off capability be required.

General Cargo (Breakbulk) Vessels: General cargo vessels (Figure C-4) are small-to-medium sized ships (compared to container vessels) that typically call on less well developed or equipped ports. They have on-board jib or boom type cranes that can be used to load or unload the ship. As the name implies, these vessels are intended to accommodate a wide variety of cargoes. Since the advent of the widespread use of containers, most of these ships are equipped with International Standards Organization lock fixtures to secure containers to the ship deck(s) and to each other. If necessary, containers can be lifted on and off these ships by using four-legged slings between the corners of the container and the hook of the crane. Because of the versatility of these vessels, casks configured for containerized or free-standing transport can be accommodated.

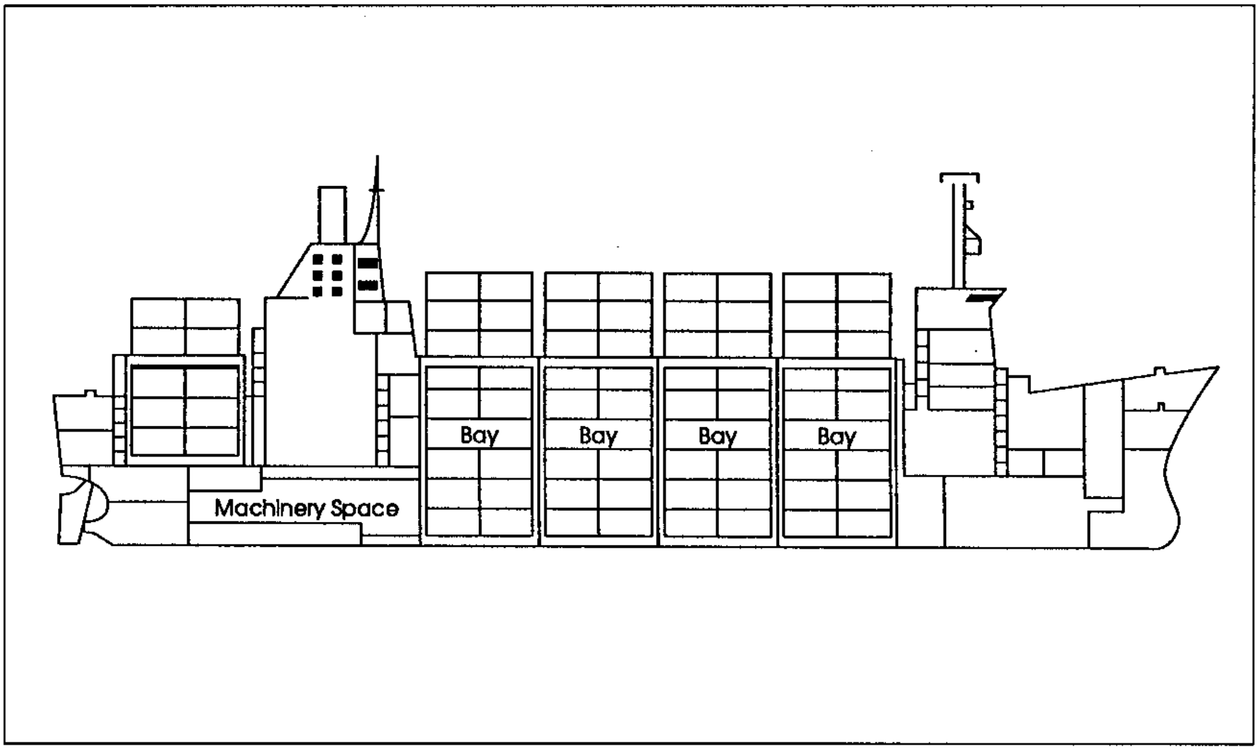


Figure C-2 Container Vessel

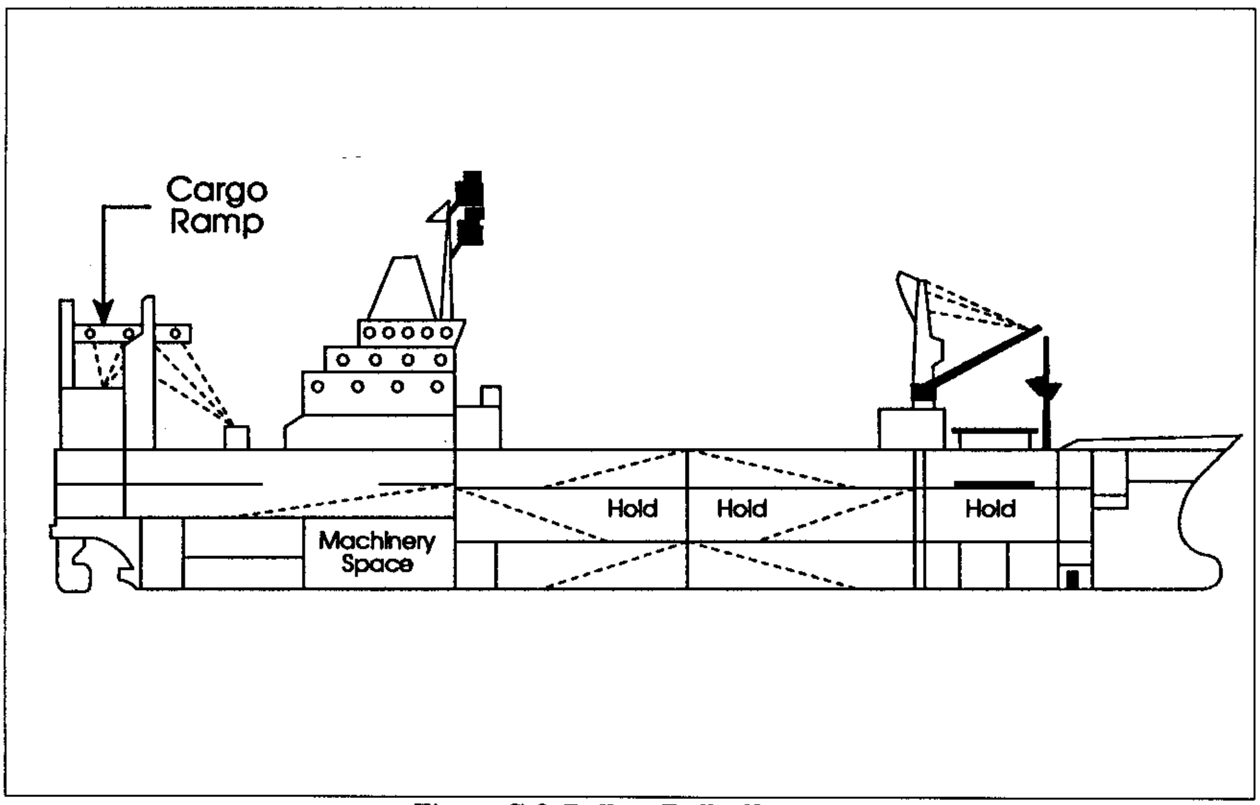


Figure C-3 Roll-on/Roll-off Vessel

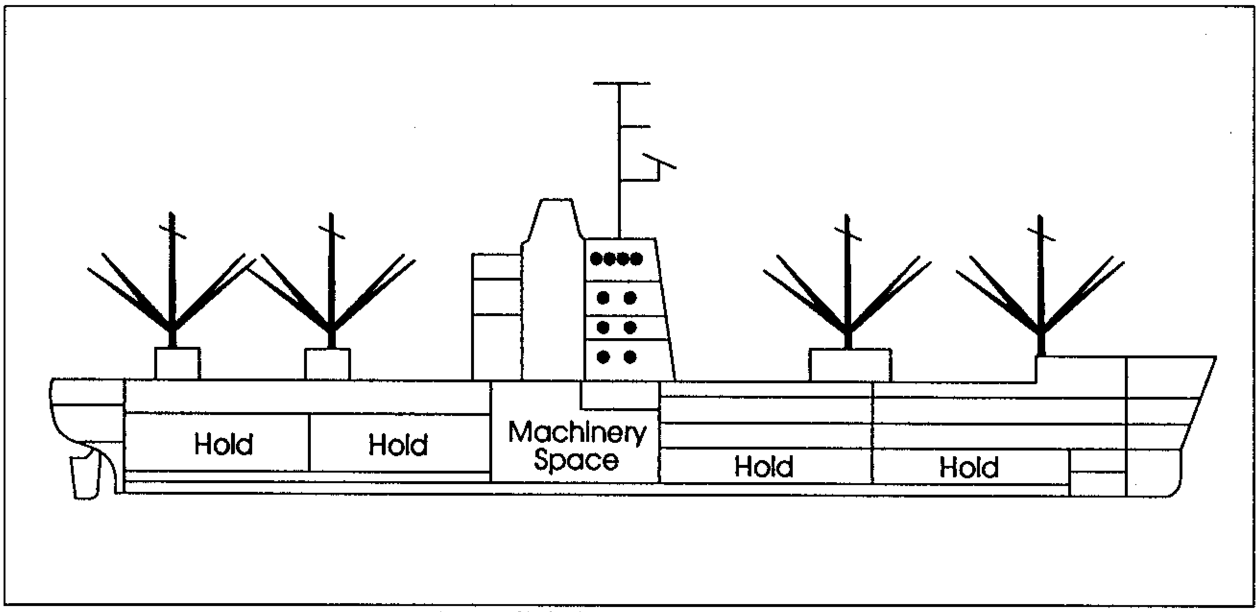


Figure C-4 General Cargo Vessel

Free-standing casks would be palletized for transport on a general cargo vessel. For stowage, the pallet would be lashed to the vessel hold or deck using conventional chains or binders. Pallets do not have standard tiedown fixtures, so there is wide variability in the specific tiedown requirements for each pallet design. Also, there is variability in provisions for lifting the pallet. The standard tiedown configuration of containers eliminates much of this variability. Consequently, containerized cask handling has resulted in an increase in the use of this configuration for the shipment of casks, and there has been a significant reduction in the number of casks shipped in the free-standing configuration.

General cargo ships have been routinely available for chartered shipment of containerized casks containing spent nuclear fuel from any number of U.S. or foreign ship lines. Because there are a comparatively small number of casks that are available for use, chartered small general cargo vessels are an option to scheduled service.

Purpose-Built Vessels: Purpose-built vessels, as used here, are those vessels specifically designed to transport spent nuclear fuel casks (Figure C-5). These vessels are not used for the transport of any other cargo and they operate as dedicated vessels. Casks are loaded directly into the holds of the vessel because the cargo compartments contain the hardware needed to mate with the tiedown fixtures of the cask. If the vessel has no crane, dockside cranes are used for loading and unloading. The cargo compartments are typically intended to handle a specific cask, and other casks cannot be used without modification to the tiedown mechanisms. For the relatively efficient transport of spent nuclear fuel, the casks normally used are very large. They are intended for the transport of power reactor spent nuclear fuel, and have a loaded weight on the order of 90 to 115 metric tons (99 to 126.5 tons). Commercial docks are not normally used, but most could be without significant problems.

The vessels have double bottoms and hulls, watertight compartments, and collision damage resisting structures within the hull. The vessel crew is trained in the handling of the cargo and in emergency response. These vessels also incorporate security features and satellite tracking systems.

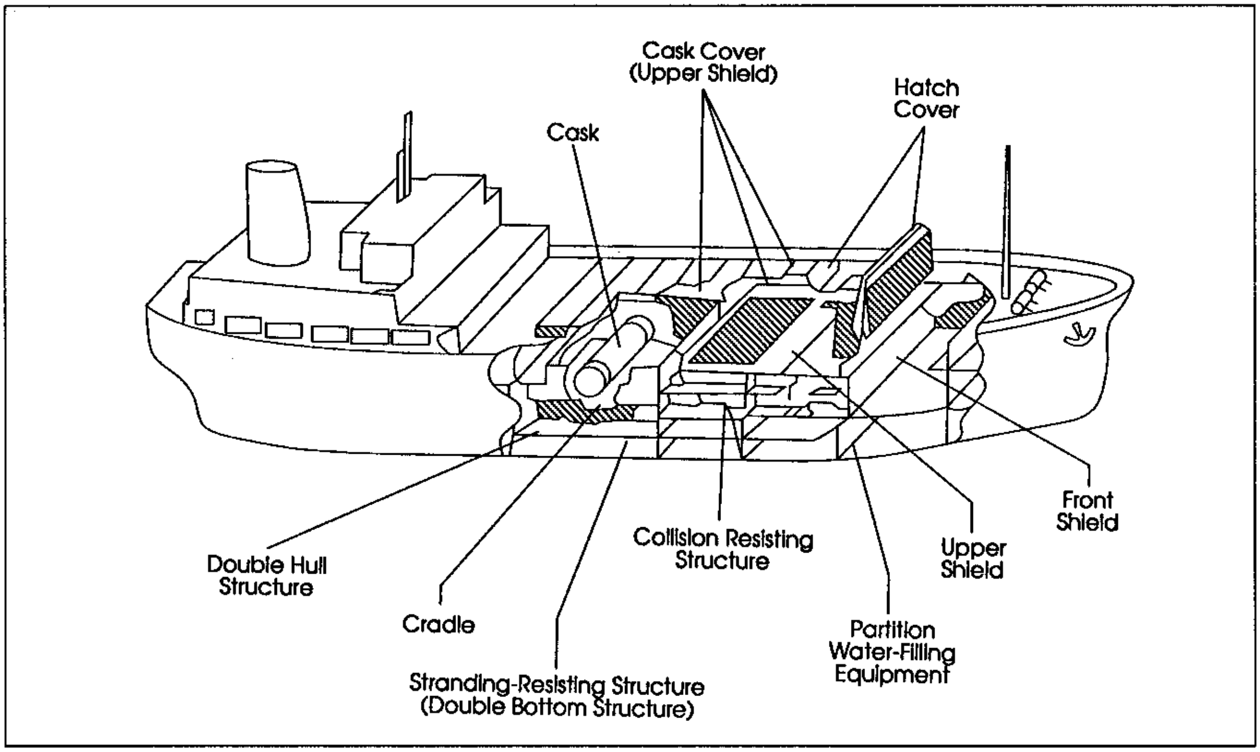


Figure C-5 Purpose-Built Ship

At present, purpose-built vessels are operated by Nuclear Transport Services of Japan, by the Swedish Nuclear Fuel and Waste Management Company, and by British Nuclear Fuels, Limited. They are used to move spent nuclear fuel from operating nuclear power plants to spent nuclear fuel reprocessing facilities operated by Cogema and British Nuclear Fuels, Limited; or, in the case of Sweden, to the repository in Forsmark. There are no U.S.-owned purpose-built vessels for spent nuclear fuel transport.

C.3.2 Identification of Routes

The foreign research reactor spent nuclear fuel that might be transported by sea under the proposed action could originate from 40 different countries. For calculation of shipping distances to the United States, shipping routes were selected to represent the transport of the fuel from a convenient port in the country of origin (for land-locked nations a port near the country of origin was selected) to both an East Coast and a West Coast U.S. port. Norfolk, VA, and Los Angeles, CA, were selected as the two port cities for use in determining a representative distance from the country of origin to the East and West Coasts of the United States. These distances were then combined to generate an average shipping distance between the country of origin and the United States. By using a city on both coasts of the United States to determine an average distance between ports, the analysis considers the possibility that shipments of foreign research reactor spent nuclear fuel would not necessarily be made to the closest U.S. port and, in fact, may be shipped to the "opposite" coast.

Table C-1 is a compilation of the distances for shipments from each of the countries that may participate in this program (except Canada) to the ports on both U.S. coasts. All route distances were obtained by using normal shipping lanes (DMA, 1991). For some of the shipments that might be received at the "opposite" U.S. coast port, the use of the Panama Canal was assumed. Other than the shipping requirements

Table C-1 Voyage Data

Country of Origin	*Distance East (nautical miles)*	*Distance West (nautical miles)*	*Average Distance (nautical miles)*	*Voyage Duration (days)*	*Number of Casks*	*Number of Voyages*
Argentina	5,824	7,265	6,545	21.2	9	5
Australia	12,728	6,511	9,620	29.7	9	5
Austria	5,026	8,955	6,991	22.9	8	4
Bangladesh	10,017	9,384	9,701	31.0	3	2
Belgium	3,582	7,782	5,682	19.3	59	30
Brazil	4,723	8,109	6,416	20.8	8	4
Chile	4,438	4,808	4,623	16.3	2	1
Colombia	2,174	3,265	2,720	11.1	1	1
Denmark	3,990	8,190	6,090	20.4	22	11
France	3,181	7,287	5,234	18.0	149	75
Finland	4,453	8,653	6,553	21.7	6	3
Germany	3,919	8,119	6,019	20.2	61	31
Greece	4,685	8,614	6,650	22.0	8	4
Indonesia	10,566	8,392	9,479	30.3	14	7
Iran	12,013	11,783	11,898	36.6	1	1
Israel	5,366	9,295	7,331	23.9	6	3
Italy	4,336	8,265	6,301	21.0	18	9
Jamaica	1,279	3,507	2,393	10.2	1	1
Japan	9,504	4,839	7,172	23.4	110	55
Korea (South)	10,480	5,229	7,855	25.3	18	9
Malaysia	10,417	7,867	9,142	28.9	3	2
Mexico	1,772	1,501	1,637	7.6	6	3
The Netherlands	3,582	7,782	5,682	19.3	49	25
Pakistan	11,460	10,749	11,105	34.4	3	2
Peru	3,172	3,655	3,414	13.0	1	1
Philippines	11,169	6,530	8,850	28.1	6	3
Portugal	3,129	7,550	5,340	18.3	3	2
Romania	5,353	9,282	7,318	23.8	48	24
Slovenia	4,172	8,372	6,272	20.9	13	7
South Africa	6,790	9,385	8,088	26.0	2	1
Spain	3,303	7,564	5,434	18.6	1	1
Sweden	4,331	8,531	6,431	21.4	37	19
Switzerland	5,026	8,955	6,991	22.9	5	3
Taiwan	11,732	7,093	9,413	29.7	9	5
Thailand	13,169	7,775	10,472	33.1	5	3
Turkey	5,002	8,931	6,967	22.9	4	2
United Kingdom	3,101	7,301	5,201	18.5	4	2
Uruguay	5,710	7,171	6,441	20.9	1	1
Venezuela	1,687	3,757	2,722	11.1	4	2
Zaire	5,864	8,583	7,224	23.6	4	2
Totals					**721**	**371**
Average				**21.3**		

Distance East - Distance in nautical miles from country of origin to Norfolk, Virginia
Distance West - Distance in nautical miles from country of origin to Los Angeles, California
Average Distance - Distance in nautical miles from country of origin to both U.S. ports
Voyage Duration - Average distance divided by 15 knots per hour plus additional days for busy way points (i.e., Panama Canal) and three days for additional stops
Number of Casks - Total casks from country of origin
Number of Voyages - Number of trips required assuming two casks per voyage

applicable to the entire journey, there are no known restrictions for spent nuclear fuel passing through either the Suez or Panama Canals. Figure C-6 provides a representation of the shipping routes selected for these shipments, although other normal shipping routes may be used.

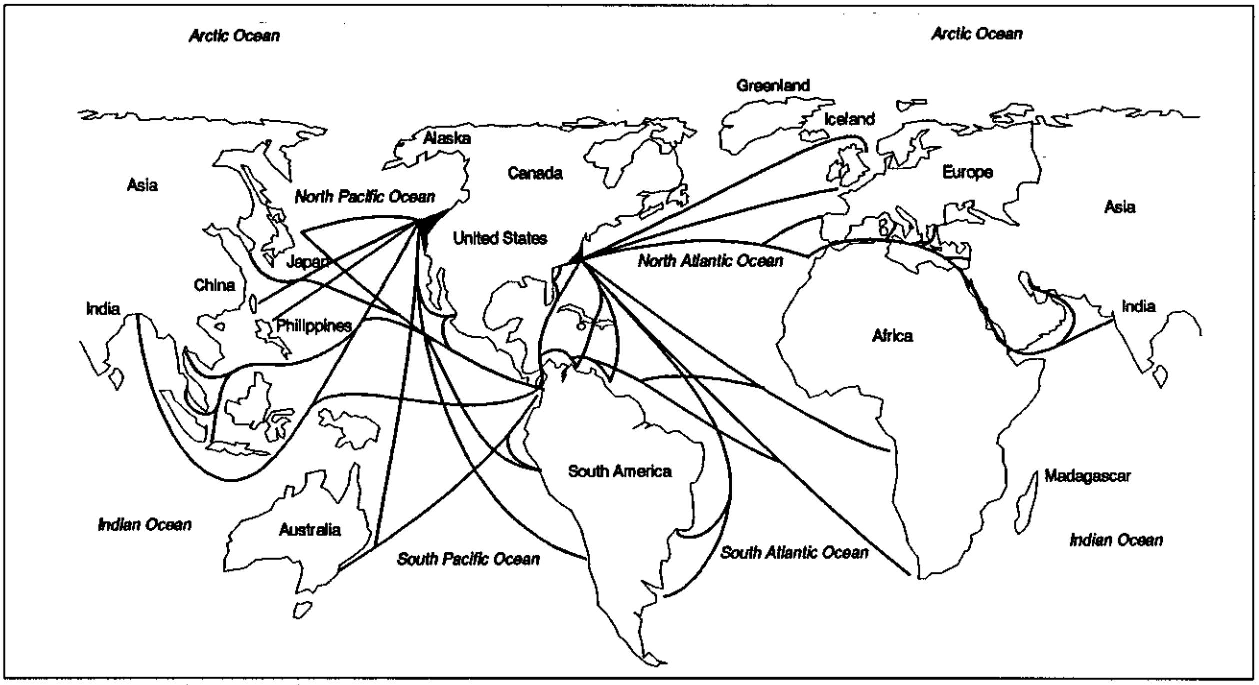


Figure C-6 Representative Shipping Routes for Foreign Research Reactor Spent Nuclear Fuel

C.4 Incident-Free Impacts: Methods and Results

C.4.1 Incident-Free Risk Assessment Methodology

External radiation from an intact transportation cask must be below specified limits that control the exposure of the handling personnel and general public. The U.S. limits are set forth in 49 CFR 173. The limit of interest established therein is 10 mrem per hour at any point 2 m (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle. This limit is associated with an "exclusive-use" shipment, which is a shipment in which no other cargo is loaded in the container used for the foreign research reactor spent nuclear fuel transportation cask and the container is not off-loaded and restowed in transit, except as directed by the shipper. This does not mean that the vessel is used exclusively for foreign research reactor spent nuclear fuel. All shipments within this program are expected to fall within this category.

In general, much of the foreign research reactor spent nuclear fuel to be received will have been out of the reactor for a significant amount of time prior to shipment, resulting in external dose rates much less than the regulatory limit. Past shipments of research reactor fuel have not approached the 49 CFR 173 limit (many, in fact, had dose rates of much less than 1 mrem per hour at 1 m). Due to the scope of this program and the possibility that some of the spent nuclear fuel may be shipped with shorter "cooldown" times than previous shipments, an analysis using typical historical dose rates may not be fully representative of all shipments. Therefore, the analysis has been performed assuming a dose rate (1) at the above-cited regulatory limit, and (2) derived from measurements taken during earlier foreign research reactor spent nuclear fuel shipments. Appendix F, Section F.5, provides a discussion of the development of the exposure dose rate versus distance relationship for a transportation cask having a dose rate at the selected exclusive-use regulatory limit.

The application of the 10 mrem per hour at 2 m (6.6 ft) exclusive-use regulatory dose limit and the "historical" dose rates provide two significant estimates for the assumed external dose rates. The exposure derived from the use of the selected regulatory limit for the dose rate is an estimate of the maximum exposure that could result from the shipments. The estimate derived from the "historical" data is closer to an expected value for the incident-free impacts. Therefore, the results of these two analyses provide an estimate of the range of incident-free impacts from the shipment of the foreign research reactor spent nuclear fuel.

The primary impact of incident-free marine transport of spent nuclear fuel is on the crews of the ships used to carry the casks. Members of the general public and marine life would not receive any measurable dose from the spent nuclear fuel during marine transport. In addition to the protection provided by the transportation casks, further protection for the general public and marine life is provided by the location of the cask in the ship (that is, the distance from the cask to the outer surface of the ship) and the ship's structure. From the outside of the ship, the foreign research reactor spent nuclear fuel shipments would be indistinguishable from any other commercial shipment. Under incident-free conditions of transport, public exposure would be limited to the ship's crew exposure, and the ship's crew exposure is limited to crew members exposed during the loading and offloading of the casks and to crew members who, on a daily basis, inspect cargo (to ensure secure stowage) and the vessel.

The type of vessel assumed to be used for transport of the spent nuclear fuel is a U.S. crewed breakbulk vessel with services not obtained on a charter basis. Breakbulk vessels typically have a number of holds, decks within each hold for carrying cargo, and their own cargo handling equipment that could be used for loading spent nuclear fuel casks. The flexibility of these vessels may be required to pick up spent nuclear fuel at some countries, since container vessel facilities may not be available.

The spent nuclear fuel cask is assumed to be in a container for ease of handling. With this assumption, the vessel with the longest cargo handling times for containerized cargo would be a breakbulk vessel. Differences in cask handling time is the key factor contributing to the differences between the incident-free impact of shipments of foreign research reactor spent nuclear fuel on different types of vessels. (See Appendix D, Section 4, for details of handling times). Therefore, the selection of this type of vessel results in a conservative estimation of the dose to the crew during transit and will bound the estimate of crew dose for any ship type selected for transport of the spent nuclear fuel.

Two different sets of assumptions have been made to assess the incident-free impacts of the shipment of foreign research reactor spent nuclear fuel. The first set of assumptions addresses the use of regularly scheduled commercial cargo vessels for the shipments. When using regularly scheduled commercial vessels, the assumption is made that two casks per vessel will be carried on each freighter, except in cases where the number of casks from a country of origin is an odd number, which would result in one shipment of only one cask. While it is likely that in some cases more than two casks per shipment could be coordinated at the same time, it is expected that the assumption of two casks per vessel should bound the incident-free analysis. The analysis assumes that both spent nuclear fuel casks are loaded into the same hold, resulting in a dose to the crew from the first cask loaded while the second cask is being loaded. This results in the crew being exposed from two sources at the same time for loading or unloading one of the two casks. Should more than two spent nuclear fuel casks be shipped on the same vessel, it has been assumed that the cargo loading would be limited to two spent nuclear fuel casks per hold. The crew would not receive any additional dose from the third, fourth, etc., cask while engaged in activities in the hold with the first two spent nuclear fuel casks. The radiological exposure to the crew for a shipment of many casks would be equivalent to the radiological exposure due to multiple shipments of fewer casks. For example, if four casks are shipped on a single vessel, the crew dose for that single shipment would be equal to the crew dose from two shipments of two casks each.

The second set of assumptions addresses the use of a chartered cargo vessel for the shipment of the foreign research reactor spent nuclear fuel. Use of a chartered vessel (either a chartered commercial freighter or a purpose-built vessel) could result in the shipment of more than two casks per voyage. Economic considerations would suggest that a larger number of casks be shipped per voyage. For this analysis it has been assumed that eight transportation casks would be shipped on a chartered vessel. Consistent with the assumption made for the regularly scheduled commercial vessel, it has been assumed that the transportation casks would be loaded two to a hold. Again, this results in doses to the crew from the first cask loaded during activities associated with the loading of the second cask in the hold.

During loading operations, both on the regularly scheduled commercial and chartered vessels, it is assumed that five members of the ship's crew (Chief Mate, Mate on Watch, Bosun, and two Seamen) will be present during loading and securing of the spent nuclear fuel casks. While longshoremen will most likely be used for the cargo handling activity, ship's crew will be present, and therefore the crew dose resulting from this activity has been included in the analysis. Table C-2 shows the crew member distances from the spent nuclear fuel shipping cask and the duration of the crew members' exposure for each crew member during the time leading up to the stowage of the cask prior to setting sail for the ocean voyage. The distances and times are based on vessel loading activities for a two-cask-per-hold shipment. The total dose (based on the selected exclusive-use regulatory limit external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) for each individual is calculated for each shipment. Since two casks are assumed to be shipped in each hold, when quantities allow, the condition exists for loading and securing of a cask to take place in the vicinity of another cask. The additional dose received by working around a cask already in the hold are accounted for in Table C-2. This was accomplished by increasing the exposure rate by a factor of 1.5 for the activities associated with securing the second cask. As listed, the estimated exposure represents the crew exposure for the regularly scheduled commercial vessel, which has been assumed to be limited to a total of two transportation casks. The exposure for each listed crew member in a chartered vessel would be four times these values, since the eight casks are assumed to be loaded into four holds.

Table C-2 Ship Crew Exposure During Loading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits)

	Tasks						
	Rail to Hold			*Secure Cask*			
Crew Member	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	*Distance[a] (m)*	*Exposure Rate[b] (mrem/hr)*	*Time (min)*	*Total Dose per Loading[c] (person-rem)*
Chief Mate	20	1	5	5.5	3.5	60	0.009
Mate on Watch	20	1	5	8	2.1	60	0.005
Bosun	20	1	5	5.5	3.5	60	0.009
Seaman (2)	20	1	5	5.5	3.5	60	0.018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 10 mrem/hr at 2 m (6.6 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded casks for activities associated with securing the second cask. The exposure rate for securing the second cask is 1.5 times the listed rate.*

While at sea, the crew dose is limited to those individuals who enter the ship's hold during transit. At all other times, the crew is shielded from the spent nuclear fuel cask by the decking and other structures of the vessel. The number of entries and inspections is a function of the voyage distance from the port of loading to the port of offloading (the U.S. port of entry for the foreign research reactor spent nuclear fuel). Since the port of offloading is unknown at this time, voyage distances were determined for each country of origin to a West and East Coast port of the United States. The average of these two distances was then calculated. Table C-1 shows the countries of origin, the number of casks, the distances to the East and West Coast ports, the average voyage distance, the days of travel, and the estimated number of casks and shipments for the basic implementation of Management Alternative 1. Because the actual shipping schedule is unknown, the average annual number of shipments was estimated. The length of a voyage was determined by assuming that the vessel would have an average speed of 15 knots for the entire duration of the voyage. In addition, intermediate port stops would be made, and additional travel time was added to account for portions of the voyage during which the vessel would not be expected to have a speed of 15 knots, (i.e., passage through busy locations, such as the Panama Canal).

Once a day while at sea or in port, the Chief Mate, the Bosun, and an Engineer are assumed to enter each cargo hold to inspect the bilges and verify the lashings for the containers. Table C-3 describes the times required for these activities, the distances from the casks during the activity, and doses received from the casks during the activity (based on the selected exclusive-use limit of external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) for each of these individuals. The total dose due to inspection activities is a function of the voyage duration and the number of holds that contain foreign research reactor spent nuclear fuel casks.

Table C-3 Ship Crew Exposure Per Hold During At-Sea Inspections (Based on Regulatory Dose Limits)

Crew Member	Distance[a] (m)	Dose Rate[b] (mrem/hr)	Time (min)	Dose per Daily Inspection[c] (mrem)
Chief Mate	5.5	7.0	20	2.3
Bosun	5.5	7.0	20	2.3
Engineer	5.5	7.0	20	2.3

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Dose rate includes the sum of the effect of two casks in hold.*

[c] *For a ship carrying two casks on a voyage duration of 21 days, the daily inspection dose to a crew member would total 48.3 mrem.*

In the analysis, two possible routes for the shipment of the spent nuclear fuel are considered. In the first, when a regularly scheduled commercial vessel is used, two intermediate port stops are assumed to add three additional days to the voyage, and therefore three additional hold inspections. The possibility of the ship having intermediate port stops must be considered in the event that a regularly scheduled commercial vessel is used for the shipment of the foreign research reactor spent nuclear fuel since the shipment is being made as part of a commercial cargo shipment. Such shipments are not limited to a single port of call. Based on the information provided in Table C-1, the average duration of a voyage would be 21 days, which includes three days for intermediate port calls. The second route accounts for using chartered ships or regularly scheduled commercial ships for which the first port of call is the port of entry for the foreign research reactor spent nuclear fuel. For this route, no intermediate port stops are included, so the travel times listed in Table C-1 were reduced by three days, making the average duration of the voyage approximately 18 days. If a ship carrying foreign research reactor spent nuclear fuel were to encounter mechanical problems or extreme weather and was forced to make an unscheduled port call, the incident-free radiation exposure to the ship's inspection crew would slightly increase as a result of the additional duration of the voyage. People in the refuge port would not receive any exposure because the foreign research reactor spent nuclear fuel would remain on the ship and would not be handled.

Once at the port of entry, all casks of the spent nuclear fuel would be off loaded. Table C-4 describes the estimated dose (based on the selected exclusive-use limit of an external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container) received by crew members involved in the offloading activities associated with the offloading of a single hold, that is, two casks. These doses are the same as those received during the loading phase of the transport activity. Once the spent nuclear fuel cask is over the rail of the ship, the ship's crew would not be in close proximity to it. As a result, no ship crew personnel are assumed to be involved with any of the activities associated with disengaging the spent nuclear fuel container from the handling gear or in securing the container to any transport vehicle used to move the container off the pier.

Table C-4 Ship Crew Exposure During Offloading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits)

	Tasks						Total Dose per Loading[c] (person-rem)
	Rail to Hold			Secure Cask			
Crew Member	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	
Chief Mate	20	1	5	5.5	3.5	60	0.009
Mate on Watch	20	1	5	8	2.1	60	0.005
Bosun	20	1	5	5.5	3.5	60	0.009
Seaman (2)	20	1	5	5.5	3.5	60	0.018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 10 mrem/hr at 2 m (6.6 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded casks for activities associated with the first cask. The exposure rate for securing the first cask is 1.5 times the listed rate.*

Tables C-5 and C-6 summarize the total crew doses for the shipment activities on a per shipment basis, annually, and for all of the shipments in the program. The maximum individual and total population doses are based on the selected exclusive-use regulatory limit external dose rate of 10 mrem per hour at 2 m or 6.6 ft from the surface of the container. Table C-5 summarizes the crew doses if regularly scheduled commercial vessels were used for all foreign research reactor spent nuclear fuel shipments. Table C-6 summarizes the crew doses if chartered vessels were used for all foreign research reactor spent nuclear fuel shipments. The reduction in the program crew-doses for the dedicated vessels is a result of the reduced transit time associated with the chartered vessels due to the fact that they do not make intermediate port calls. In situations where the services of a ship are obtained on a non-exclusive-use basis, the maximum allowable annual dose to a member of the ship's crew would be 100 mrem per year [based on Nuclear Regulatory Commission (NRC) and U.S. DOE limits on the exposure of members of the public].

As shown in Table C-5, the maximum individual dose per shipment on a regularly scheduled commercial vessel is 66 mrem to the Chief Mate and Bosun, a dose well below the 100 mrem per year limit. If the assumption was made that the same vessel and crew was used for as many shipments as possible in one year, the maximum individual dose to a crew member would be approximately 600 mrem. This assumes

Table C-5 Total Regularly Scheduled Commercial Ship's Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits and Assuming Intermediate Port Stops)

Crew Member	Maximum Individual Exposure per Trip (mrem)	Maximum Individual Exposure per Year[a] (mrem)	Program Dose Total (All Ships' Crews) (person-rem)
Chief Mate	66	599	24.8
Mate on Watch	11	98	4.0
Bosun	66	599	24.8
Seaman (2)	18	158	13.1
Engineer	49	441	18.2
Total			**84.9**

[a] *Exposure per year based on nine voyages per year, two casks per voyage.*

Table C-6 Total Chartered Ship's Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Regulatory Dose Limits and Assuming No Intermediate Port Stops)

Crew Member	Maximum Individual Exposure per Trip (mrem)	Maximum Individual Exposure per Year[a] (mrem)	Program Dose Total (All Ships' Crew) (person-rem)
Chief Mate	238	1,668	21.7
Mate on Watch	43	303	3.9
Bosun	238	1,668	21.7
Seaman (2)	70	492	12.8
Engineer	168	1,176	15.3
Total			**75.4**

[a] *Exposure per year based on seven trips per year, eight casks per voyage (two casks per hold).*

nine trips per year based on the average voyage length of all shipments and results in the ships' crew being exposed to the foreign research reactor spent nuclear fuel shipments for 189 days a year. Since travel time to a port of loading would be required, and most ship crews are rotated on a three or six month basis, the assumption of nine trips should bound the dose for any individual members of dedicated crews, even when trips are shorter than the assumed average of 21 days. The annual dose of approximately 600 mrem exceeds the 100 mrem annual limit for a member of the general public, and would therefore require mitigation. See the end of this section for a discussion of mitigation.

Due to the larger number of casks on a chartered vessel, the largest annual dose to a crew member is estimated to be approximately 1,668 mrem (approximately 1.7 rem). This is based on an estimated exposure of 238 mrem per voyage and seven voyages per year. Seven voyages per year using a chartered vessel is sufficient to ship all transportation casks to be shipped in an average year. It has been assumed that the 721 shipments would be made over a 13-year period. The exposure total for the marine transport portion of the program can be expressed as the number of LCFs that are calculated to result from doses received during the policy period, if the basic implementation of Management Alternative 1 of the proposed action were adopted. For a regularly scheduled commercial vessel, the exposure of approximately 84.9 person-rem translates to 0.034 LCFs. The total exposure associated with the shipment of the foreign research reactor spent nuclear fuel on a chartered vessel, approximately 76.4 person-rem, translates into 0.031 LCFs.

Use of a chartered vessel results in a reduction of approximately ten percent in the total population exposure and corresponding risk to the ships' crews under the basic implementation of Management Alternative 1. This difference is due to the shorter voyage duration when a chartered vessel is used. From Tables C-2 through C-4, it is apparent that the largest doses to the ship's crew are a result of the daily inspection of the cargo holds. The three day reduction in the voyage duration (gained when a chartered vessel is used) reduces the dose received from the daily inspections and results in the ten percent difference between the use of regularly scheduled commercial and chartered vessels.

Tables C-7 through C-11 present the results of the above analysis with one change. The exposure and crew doses are calculated based on the "historical" external dose rate data developed from measurements taken during earlier shipments of research reactor spent nuclear fuel (a dose rate of 2.25 mrem per hour at 1 m or 3.3 ft from the surface of the shipping cask, which is equivalent to 1 mrem per hour at 2 m or 6.6 ft from the cask surface). See Appendix F, Section F.5 for the data used to derive this historical dose rate. Although this "historical" data are based on distance from the surface of the cask, it has conservatively been assumed in this analysis that this dose rate represents the dose at distances from the surface of the container in which the cask is shipped. This set of calculations was performed in order to provide additional perspective about the risks associated with the foreign research reactor spent nuclear fuel program. Use of the exclusive-use regulatory limit for the external dose rate ensures that the estimates discussed previously are upper bounds on the potential risks to the ship's crew from incident-free transport of the spent nuclear fuel. Use of the historical data provides an estimate that is closer to the expected risks associated with the shipment of all of the foreign research reactor spent nuclear fuel. Although the exact external dose rates cannot be determined in advance for all shipments, most should be similar to those for shipments made in the past. Therefore, the "historical" external dose rates should be a more accurate prediction of the risks resulting from the shipment of all 721 casks.

In this analysis, all other assumptions regarding voyage length, crew activity (time and distance from the spent nuclear fuel cask), number of shipments, and the assumptions made to estimate annual doses remained the same as in the analysis performed using the external dose rates derived from the exclusive-use regulatory limit of 10 mrem per hour at 2 m (6.6 ft) from the surface of the shipping container.

Using the historic dose rates, the maximum dose to an individual per regularly scheduled commercial vessel shipment would be 6.6 mrem, and the annual maximum individual dose would be 60 mrem (this dose is calculated assuming that the same crew member is involved in nine separate voyages transporting two spent nuclear fuel casks each during a single year). These doses are an order of magnitude lower than the corresponding doses calculated using the exclusive-use regulatory external dose rates. The calculated maximum individual dose is well below the maximum allowable annual dose to a member of the public of 100 mrem.

Use of a chartered vessel for the shipments, versus the use of a regularly scheduled commercial vessel, would result in a ten percent reduction in the total ships' crews doses. The use of a chartered vessel would result in annual exposure at slightly less than twice the public dose limits for exposure to radiation established by both DOE and NRC (100 mrem per year).

The dose total for the marine transport portion of the entire program can be expressed as the number of LCFs that are calculated to result from exposures of that size. For a regularly scheduled commercial vessel a total exposure of approximately 8.5 person-rem translates to 0.0034 LCFs. The total calculated exposure associated with the shipment of the foreign research reactor spent nuclear fuel on a chartered vessel, approximately 7.6 person-rem, translates into 0.0030 LCFs.

The results of these analyses indicate that, in some circumstances, some individual crew members could receive doses that exceed the limit established by DOE and the NRC for exposure of a member of the public, especially when the dose rate from the casks are assumed to be at the regulatory limit. It is anticipated that for most shipments, the external dose rate for the loaded transportation case would be near

Table C-7 Ship Crew Exposure During Loading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Historical Cask Dose Rates)

Crew Member	Tasks						Total Dose per Loading[c] (person-rem)
	Rail to Hold			Secure Cask			
	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	
Chief Mate	20	0.1	5	5.5	0.35	60	0.0009
Mate on Watch	20	0.1	5	8	0.21	60	0.0005
Bosun	20	0.1	5	5.5	0.35	60	0.0009
Seaman (2)	20	0.1	5	5.5	0.35	60	0.0018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 2.25 mrem/hr at 1 m (3.3 ft) from the shipping container surface.*

[c] *Includes the exposure from the first loaded cask for activities associated with securing the second cask. The exposure rate for securing the second cask is 1.5 times the listed number.*

Table C-8 Ship Crew Exposure Per Hold During At-Sea Inspections (Based on Historical Cask Dose Rates)

Crew Member	Distance[a] (m)	Dose Rate[b] (mrem/hr)	Time (min)	Dose per Daily Inspection[c] (mrem)
Chief Mate	5.5	0.7	20	0.23
Bosun	5.5	0.7	20	0.23
Engineer	5.5	0.7	20	0.23

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Includes the effect of two casks in the hold.*

[c] *For a ship carrying two casks on a voyage duration of 21 days, the total dose to a crew member conducting daily inspections would be estimated at 4.8 mrem.*

Table C-9 Ship Crew Exposure During Offloading of a Hold Containing Two Foreign Research Reactor Spent Nuclear Fuel Casks (Based on Historical Cask Dose Rates)

Crew Member	Tasks						Total Dose per Loading[c] (person-rem)
	Rail to Hold			Secure Cask			
	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	Distance[a] (m)	Exposure Rate[b] (mrem/hr)	Time (min)	
Chief Mate	20	0.1	5	5.5	0.35	60	0.0009
Mate on Watch	20	0.1	5	8	0.21	60	0.0005
Bosun	20	0.1	5	5.5	0.35	60	0.0009
Seaman (2)	20	0.1	5	5.5	0.35	60	0.0018

[a] *Distance is the average distance of the crew member from the spent nuclear fuel cask during the entire duration of that activity.*

[b] *Exposure rate is calculated based on 2.25 mrem/hr at 1 m (3.3 ft) from the shipping container surface.*

[c] *Includes the exposure from the last off loaded cask during activities associated with the first off loaded cask. The exposure rate for securing the first cask is 1.5 times the listed rate.*

Table C-10 Total Regularly Scheduled Commercial Ships Crew Exposure for Marine Transport of Foreign Research Reactor Spent Nuclear Fuel Casks Assuming Intermediate Port Stops (Based on Historical Cask Dose Rates)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Dose Total (All Ships' Crew) (person-rem)*
Chief Mate	6.6	60	2.5
Mate on Watch	1.1	10	0.4
Bosun	6.6	60	2.5
Seaman (2)	1.8	16	1.3
Engineer	4.9	44	1.8
Total			**8.5**

[a] *Exposure per year based on nine trips per year.*

Table C-11 Total Chartered Ships Crew Exposure for Marine Transport of Foreign Research Reactor Spent Fuel Casks Assuming No Intermediate Port Stops (Based on Historical Cask Dose Rates)

Crew Member	*Maximum Individual Exposure per Trip (mrem)*	*Maximum Individual Exposure per Year[a] (mrem)*	*Dose Total (All Ships' Crew) (person-rem)*
Chief Mate	24	167	2.2
Mate on Watch	4.3	30	0.4
Bosun	24	167	2.2
Seaman (2)	7.0	49	1.3
Engineer	17	118	1.5
Total			**7.6**

[a] *Exposure per year based on seven trips per year.*

the historic dose rates, which would not cause any personnel to exceed radiation exposure limits for the public. However, the existence of some shipments with external dose rates closer to the exclusive-use regulatory limit suggests that DOE should provide a means to assure that individual crew members do not receive doses in excess of the public dose limits. As a minimum, the program should establish administrative procedures that will maintain records of the dose rates associated with each shipment and the ports of departure and entry for the shipment. The measurement of interest for the record keeping would be the external dose rates outside the container, which houses the transportation cask, since the crew does not enter the container. (It should be noted that the analysis using the historical data did not consider the reduction in external dose rate due to the distance from the cask to the container surrounding the transportation cask.) These measurements can be used to identify shipments that would result in crew exposures above those calculated based on the historical spent nuclear fuel transportation external dose rate. By tracking this information, DOE would be able to identify if and when additional precautions to reduce individual exposures should be taken (i.e., restricting the use of crew members who are near the annual dose limit from further shipments that year). DOE would also include a clause in the contract for shipment of the foreign research reactor spent nuclear fuel requiring other crew members be used if any crew member approaches a 100 mrem dose in any year.

C.4.2 Incident-Free Marine Impacts of Policy Alternatives

Two implementation subalternatives to Management Alternative 1 and one subalternative under Management Alternative 2 of the proposed action were identified that could impact the incident-free marine risk calculations that were performed for the basic implementation (Chapter 2 describes the alternatives and subalternatives of Management Alternative 1 and the subalternatives of Management

Alternative 2). The implementation subalternative of accepting spent nuclear fuel only from developing countries would result in a reduction in the amount of spent nuclear fuel transported by ship. Table C-12 lists the countries that are considered developing countries and the number of shipments that would be required to transport their spent nuclear fuel to the United States.

Table C-12 Voyage Data for Acceptance of Foreign Research Reactor Spent Nuclear Fuel from Developing Nations Only

Port	*Voyage Duration (days)*	*Number of Casks*	*Number of Trips*
Argentina	21.2	9	5
Bangladesh	31.0	3	2
Brazil	20.8	8	4
Chile	16.3	2	1
Colombia	11.1	1	1
Greece	22.0	8	4
Indonesia	30.3	14	7
Iran	36.6	1	1
Jamaica	10.2	1	1
Korea (South)	25.3	18	9
Malaysia	28.9	3	2
Mexico	7.6	6	3
Pakistan	34.4	3	2
Peru	13.0	1	1
Philippines	28.1	6	3
Portugal	18.3	3	2
Romania	23.8	48	24
Slovenia	20.9	13	7
South Africa	26.0	2	1
Thailand	33.1	5	3
Turkey	22.9	4	2
Uruguay	20.9	1	1
Venezuela	11.1	4	2
Zaire	23.6	4	2
Totals		**168**	**90**
Average	**23**		

Under the implementation subalternative of using a policy duration of five years for the acceptance of foreign research reactor spent nuclear fuel, the number of transportation casks of foreign research reactor spent nuclear fuel requiring ocean transport would be reduced to 586. Appendix B presents the derivation of the total number of shipments (ocean transport plus land transport from Canada) estimated in this alternative.

Subalternative 1b (overseas reprocessing) under Management Alternative 2 also has the capability to impact the results of the incident-free marine risk analysis since it involves shipment of the vitrified waste to a storage facility in the United States. Under this subalternative to Management Alternative 2, eight transportation cask shipments of vitrified waste would be made to the United States.

In addition, a Hybrid Alternative was analyzed. In the Hybrid Alternative, those countries (for this option, assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom) that have the capability to store high-level waste would be encouraged to reprocess the aluminum-based research reactor spent nuclear fuel and to accept for management the resulting high-level waste. The United States would accept for management the research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability, and all TRIGA fuel. This Hybrid Alternative includes all countries identified in Table C-1 except for those seven nations just listed. Under this Hybrid Alternative, 452 shipments of spent nuclear fuel are assumed to be sent to the United States, excluding shipments of Canadian origin.

The incident-free marine risks associated with the two implementation subalternatives of Management Alternative 1 and the subalternative of Management Alternative 2 are discussed in the following sections.

Management Alternative 1, Implementation Subalternative 1a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only from Developing Countries: This implementation subalternative of Management Alternative 1 would result in the shipment of 168 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact of the basic implementation of Management Alternative 1 have been used in the analysis of this implementation subalternative. This implementation subalternative has been analyzed using the "exclusive-use" shipment regulatory transportation cask external dose rates. To compare this implementation subalternative to the basic implementation, it is only necessary to perform the analysis using one estimate of the external dose rate of the transportation cask. The relationship between the calculated impact of the two implementation subalternatives using the regulatory external dose rate would be the same as that calculated using the "historical" data. Therefore, the use of the one dose rate provides a sufficient point of comparison between the two alternatives.

The assumptions that have not changed between the analysis for the basic implementation and this implementation subalternative include the following:

- The same types of vessels should be available for use, so, the option for using chartered or regularly scheduled commercial vessels was examined, and
- The activities associated with the loading of the foreign research reactor spent nuclear fuel, the daily inspections of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel do not change simply because there is a reduction in the number of shipments to be made.

The average duration of the voyages from these developing countries to the United States is slightly longer than the average for the voyages associated with the basic implementation. As shown in Table C-12, the average duration is 23 days (for a regularly scheduled commercial vessel) versus the 21 days in the basic implementation. For a chartered vessel, the voyage duration is three days less (i.e., 20 days). The longer average voyage duration results in an increase in the total of the daily inspection-related crew doses of approximately 4.6 mrem per crew member involved in the inspection. The inspection dose for a 23-day voyage would be 52.9 mrem (2.3 mrem times 23 days) per inspector.

The population dose to the ship's crew, per voyage, can be derived from the data contained in Tables C-5 and C-6. Incorporating the increase in the inspection dose into the data from Table C-5, the individual doses on a regularly scheduled commercial vessel would be 71 mrem to the Chief Mate and the Bosun, 11 mrem to the Mate on Watch, 18 mrem to each of two Seamen, and 54 mrem to the Engineer. The population (ship's crew) dose per shipment would be 242 mrem. If a chartered vessel is used (carrying eight transportation casks instead of two for the regularly scheduled commercial vessel), the corresponding doses are 257 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 187 mrem to the Engineer. The population (ship's crew) dose per shipment would be 885 mrem.

The 168 cask-shipments, requiring 90 ocean voyages using regularly scheduled commercial cargo vessels (up to 23 voyages using chartered vessels), represent approximately 24 percent of the total number of shipments in the basic implementation. The total population (ship's crew) exposure resulting from this implementation subalternative would be approximately 27 percent of the exposure calculated for the basic implementation. The difference in these two percentages is a direct result of the longer average duration of ocean crossings. The total population exposure for the implementation subalternative, assuming that regularly scheduled commercial vessels are used, would be approximately 22.0 person-rem, and would be approximately 20.3 person-rem if chartered vessels are used. These population exposures translate into a risk to the ship's crew of 0.0091 LCF and 0.0081 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs for workers (ship's crew) is that a 1 rem dose equates to 0.0004 LCFs.

Management Alternative 1, Implementation Subalternative 2a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel for Five-Year Policy Duration: As stated above, this implementation subalternative results in the shipment of 586 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact of the basic implementation have been used in the analysis of this implementation subalternative. This implementation subalternative has been analyzed using the "exclusive-use" shipment regulatory transportation cask external dose rates. To compare this implementation subalternative to the basic implementation it is only necessary to perform the analysis using one external dose rate. The relationship between the calculated impact of the implementation subalternative and the basic implementation using the regulatory external dose rate would be the same as that calculated using the "historical" data. Therefore, the use of the one dose rate provides a sufficient point of comparison.

The assumptions that have not changed between the analysis for the basic implementation and this implementation subalternative include the following:

- The same types of vessels should be available for use, and the option for using chartered or regularly scheduled commercial vessels was examined;
- The average voyage duration that was used in the analysis of the incident-free marine risk for the basic implementation was used for this implementation subalternative. The 586 shipments represent approximately 81 percent of the shipments made under the basic implementation and the distribution of shipments from the different countries of origin is similar to that modeled for the basic implementation; and
- The activities associated with the loading of the foreign research reactor spent nuclear fuel transportation casks, the daily inspections of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel transportation casks do not change simply because there is a reduction in the number of shipments to be made.

Because there are no differences between the per-shipment activities in this implementation subalternative and the basic implementation, the per-voyage crew exposures will not differ from those presented in Tables C-5 and C-6 for the basic implementation. In addition, the maximum annual exposures to individual crew members will not change. The analysis has assumed a maximum number of voyages that a single crew would be involved in during a single year. Although the total number of shipments per year must increase in this alternative (an average of 73 casks must be shipped per year for eight years), no single ship's crew will be involved in more shipments than had been assumed in the analysis of the basic implementation. The annual doses presented in Tables C-5 and C-6 are applicable to this alternative as well as to the basic implementation.

The total population (ship's crew) exposure resulting from this implementation subalternative would be approximately 81 percent of exposure calculated for the basic implementation. The total population exposure for the implementation subalternative, assuming that regularly scheduled commercial vessels are used, would be approximately 69 person-rem, and would be approximately 61 person-rem if chartered vessels were to be used. These population exposures translate into a risk to the ships' crew of 0.028 LCF and 0.025 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs for workers (ship's crew) is that a 1 rem dose equates to 0.0004 LCF.

Management Alternative 2, Subalternative 1b – Overseas Processing with Shipment of Waste to a U.S. Storage Facility: In this subalternative, the foreign research reactor spent nuclear fuel would be reprocessed overseas (most probably in Great Britain or France) and the waste products would be contained within a small number of vitrified waste logs. This high-level waste might be brought to the United States for storage at one of the management site facilities evaluated under the basic implementation of Management Alternative 1. Under these conditions, up to eight transportation casks containing 16 European-size canisters of vitrified waste would be shipped from Europe to the United States (see Section 4.4.2.2 for more information on the vitrification of the waste material). This analysis addresses the incident-free marine risks associated with transporting these eight casks of vitrified waste from Europe to the United States.

As with the shipment of unprocessed spent nuclear fuel, the primary impact of incident-free marine shipping of the vitrified waste is upon the crews of the ships used to carry the casks. Most of the assumptions used in the analysis of the crew exposure to the spent nuclear fuel (see Section C.4.1 of this appendix) have been used to analyze the impact of the shipment of vitrified waste. The crew exposure due to loading and offloading activities have been considered, but the primary contribution to the crew dose comes from the daily cargo inspection activities. The inspection activities on the ship carrying the vitrified waste have been modeled in the same manner as the inspections aboard the vessels carrying the spent nuclear fuel. Three crew members have been modeled as performing the inspections, and the same three crew members are assumed to perform this task for the entire voyage. For the purposes of this analysis, it has been assumed that the vitrified waste will be transported on a chartered vessel, there will be no intermediate port calls, and the shipment will originate in Europe. Because there are no intermediate port calls and the shipments originate in Europe, the voyage duration is estimated to be 15 days. This estimate is based on the average of the voyage durations for one trip from the United Kingdom to the East Coast of the United States, one to the West Coast of the United States, and the average of a trip from France to both U.S. coasts. The assumption that there are no intermediate port calls reduces the average duration of each of these trips by three days from the estimates presented in Table C-1.

Little information is available on the casks that might be used to transport the vitrified waste. Therefore, the assumption has been made that the exposure to the crew will be limited to the exclusive-use regulatory limit (10 CFR 71) of 10 mrem per hour at 2 m (6.6 ft) from the surface of the container. No attempt was made to extrapolate limited historical data to determine crew incident-free impacts from any other exposure rate other than the limit set forth in NRC and DOE regulations.

It has been assumed that two casks are being transported as part of a single shipment. This assumption results in additional exposure to the crew members due to exposure to two radiation fields during all activities which bring crew members into the vicinity of the transportation casks. Should all of the casks be shipped at once, this assumption is equivalent to assuming that this single shipment is made with two casks per hold on the vessel. The crew risk would be the same for this single (eight cask) shipment as for the four shipments with two casks per vessel.

Based on the assumptions outlined above, the incident-free impact of the shipment of vitrified waste on the ship's crew would be slightly less per shipment than that calculated for the shipment of foreign research reactor spent nuclear fuel. The trip duration of only 15 days, versus the average duration of 18 days, for a chartered vessel in the basic implementation of Management Alternative 1 results in a reduction of the dose to each inspector, the Chief Mate, the Bosun, and the Engineer, of approximately 6.9 mrem per journey (three fewer inspections, each of which would have resulted in a dose of 2.3 mrem). The population dose to the ship's crew, per voyage, can be derived from the data contained in Table C-6. Incorporating the reduction in the inspection dose into the data from this table, the individual doses would be: 210 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 140 mrem to the ships Engineer. Per voyage, the total population dose to the ship's crew would be 0.74 person-rem.

With only eight casks to be shipped, the subalternative action could be achieved with a single shipment (the crew dose would be the same as that calculated if four shipments of two casks each were made). The population exposure results in a risk to the crew of 0.00030 LCF. Due to the reduced number of shipments, compared to the 721 shipments of spent nuclear fuel in the basic implementation of Management Alternative 1, the marine incident-free risk to the crew is approximately two orders of magnitude lower than that calculated for the basic implementation.

Management Alternative 3 – Combination of Components of Management Alternative 1 and 2 (Hybrid Alternative): Under the Hybrid Alternative, the United States would accept foreign research reactor spent nuclear fuel from countries without high-level waste storage capability. This Hybrid Alternative could result in the shipment of 452 casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free marine impact for the basic implementation of Management Alternative 1 have been used in the analysis of this Hybrid Alternative. This alternative has been analyzed using the selected "exclusive-use" regulatory dose limit for the shipment of spent nuclear fuel casks.

Included in the assumptions that have not changed between the analysis for the basic implementation and this alternative are the following:

- The same types of vessels should be available for use under this Hybrid Alternative, the option for using chartered or regularly scheduled commercial vessels was examined, and

- The activities associated with the loading of the foreign research reactor spent nuclear fuel, the daily inspection of the cargo during the voyage, and the offloading of the foreign research reactor spent nuclear fuel do not change simply because there is a reduction in the number of shipments to be made.

The average duration of the voyages from the countries without high-level waste storage capability to the United States is slightly longer than the average for the voyages associated with the basic implementation. Using the data in Table C-12, and eliminating the aluminum-based spent fuel shipments from Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom, the average voyage duration is almost 23 days (for a regularly scheduled commercial vessel) versus the 21 days for the basic implementation. For a chartered vessel, the voyage duration is three days less (i.e., almost 20 days). The longer average voyage duration results in an increase in the total of the daily inspection-related crew doses of approximately 4.6 mrem per crew member involved in the inspection. The inspection dose for a 23-day voyage would be 52.9 mrem (2.3 mrem times 23 days) per inspector.

The population dose to the ship's crew, per voyage, can be derived from the data contained in Tables C-5 and C-6. Incorporating the increase in the inspection dose into the data from Table C-5, the individual doses on a regularly scheduled commercial vessel would be 71 mrem to the Chief Mate and the Bosun, 11 mrem to the Mate on Watch, 18 mrem to each of two Seamen, and 54 mrem to the ship's Engineer. The population (ship's crew) dose per shipment would be 243 mrem. If a chartered vessel is used (carrying eight transportation casks instead of two for the regularly scheduled commercial vessel), the corresponding doses are 257 mrem to the Chief Mate and the Bosun, 43 mrem to the Mate on Watch, 70 mrem to each of two Seamen, and 187 mrem to the ship's Engineer. The population (ship's crew) dose per shipment would be 884 mrem.

The 452 cask shipments, requiring 236 ocean voyages using commercial regularly scheduled commercial cargo vessels, represent approximately 63 percent of the total number of shipments for the basic implementation. The total population (ships' crew) exposure resulting from this Hybrid Alternative would be approximately 69 percent of the exposure calculated for the basic implementation. The differences in these two percentages is a direct result of the longer average duration of ocean crossings. The total population exposure for the Hybrid Alternative, assuming that regularly scheduled commercial vessels are used, would be approximately 57.2 rem and would be approximately 52.2 rem if chartered vessels were used. These population exposures translate into a risk to the ships' crew, in terms of LCFs, of 0.024 LCF and 0.021 LCF, respectively. As discussed in Section 4.1, the relationship between a dose and LCFs is that a 1 rem dose equates to 0.0004 LCFs.

C.5 Accident Impacts: Methods and Results

C.5.1 Introduction

If the cask sinks anywhere in U.S. coastal waters, it will be recovered, regardless of depth. U.S. coastal waters in this case refers to waters within the 12-mile territorial limit. Recovery would be accomplished, even in the deepest parts of U.S. coastal waters, such as in Puget Sound, which reaches 305 meters or 1,000 feet (Encyclopedia Americana, 1991). Elsewhere in the world, if the cask sinks in coastal water (i.e., in water up to 200 m or 660 ft), every effort would be made to recover it. In deeper waters, the recovery is more problematic. As recovery, even in coastal waters, cannot be guaranteed, two scenarios need to be evaluated:

> ***Scenario A***: As the result of a maritime casualty (e.g., collision, foundering, fire), the vessel sinks in coastal waters, resulting in the submersion of the cask on the ocean floor. The cask is not retrieved. Analyses are done for two cases, (1) damaged cask, and (2) undamaged cask.

> ***Scenario B***: As the result of a maritime casualty (e.g., collision, foundering, fire), the vessel sinks in deep ocean waters, resulting in the submersion of the cask on the ocean floor. The cask is not retrieved. Analyses are done for one case only, a damaged cask, as it has been assumed that submersion in the deep ocean will damage the cask.

In 1988, the Nuclear Energy Agency of the Organization for Economic Cooperation and Development published a radiological assessment as part of a feasibility study for disposal of high-level radioactive waste into the seabed (NEA, 1988). As part of the radiological assessment, several accident scenarios were examined. In particular, a scenario involving a transportation accident at sea was examined. The results of calculations performed for the Nuclear Energy Agency radiological assessment are used here, with modification. The Nuclear Energy Agency results are based on vitrified high-level waste, which behaves differently in salt water than the metal foreign research reactor spent nuclear fuel. Also, the inventory of radioactive material in the foreign research reactor spent nuclear fuel is considerably different than the vitrified high-level waste inventory. With modifications to compensate for these differences, the Nuclear Energy Agency results were used to predict the peak individual dose and biota dose for Scenario A and Scenario B.

C.5.2 Assumptions

1. The spent nuclear fuel and cask modeled are the BR-2 fuel and the Pegase cask. Based on the information provided in Appendix B, the loaded Pegase cask contains 0.0155 metric tons of heavy metal (MTHM) (15.5 kg) of fuel (assuming the cask is loaded with BR-2 type fuel). This fuel type was selected because BR-2 fuel has the highest isotope content per unit mass of heavy metal of the three fuel types considered in this analysis. Use of the highest inventory of radionuclides establishes a conservative upper bound on the estimated dose rates from the leaching of radionuclides into the sea. This is because the dose rates are a function of the corrosion rate of spent nuclear fuel, expressed in terms of mass per unit of time, and the specific activity of the spent nuclear fuel, expressed in terms of radioactivity per unit of mass.

2. The fuel rods contain aluminum-clad metallic spent nuclear fuel elements.

3. The deep ocean model is for the South Nares Abyssal Plain.

4. Corrosion of spent nuclear fuel inside a damaged cask begins immediately; corrosion of spent nuclear fuel inside an undamaged cask begins at the time the cask fails and allows seawater to come in contact with the spent nuclear fuel.

5. Once free of the fuel matrix through corrosion, the fission products exit the failed cask without delay.

6. The corrosion rate for spent nuclear fuel elements is constant. Radionuclides are leached from the spent nuclear fuel elements at a rate proportional to the corrosion rate depending on their relative concentrations.

Data from the Nuclear Energy Agency vitrified high-level waste model and on spent nuclear fuel corrosion rates are summarized in Table C-13.

Table C-13 Data For Estimating Spent Nuclear Fuel Dose Rates From the Nuclear Energy Agency Assessments for Vitrified High-Level Waste

Parameter Description	*Value*[a]	*Source*
Corrosion Rate for Glass (α_0)	0.000036 kg/m^2 day	NEA 1988
Corrosion Rate for Aluminum-Clad Fuel (α_1)	0.0086 kg/m^2 day	Rechard 1994
Sensitivity Coefficient for Corrosion Rate (a)	0.99	NEA 1988
Undamaged Cask Peak Individual Dose	9 rem/yr	NEA 1988
Damaged Cask Peak Individual Dose	650 rem/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Fish)	3.6 rad/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Crustaceans)	3.8 rad/yr	NEA 1988
Undamaged Cask Peak Biota Dose (Mollusks)	10.0 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Fish)	29.0 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Crustaceans)	31 rad/yr	NEA 1988
Damaged Cask Peak Biota Dose (Mollusks)	660 rad/yr	NEA 1988

[a] *Dose rates are based on a total Nuclear Energy Agency program mass of 100,000 MTHM*

C.5.3 Calculational Method For Dose Rate Estimates

The calculations presented here are designed to account for two differences between the Nuclear Energy Agency radiological assessment and the radiological assessment required for this EIS. First, in the radiological assessment performed for the Nuclear Energy Agency, a vitrified glass waste form was assumed. For this EIS, aluminum-clad metal matrix fuel elements are assumed. Thus, the corrosion rate of the matrix containing the radionuclides will be different in the two cases. Second, the radiological assessment for the Nuclear Energy Agency was performed assuming reprocessed fuel equivalent to 100,000 MTHM containing a total of 10 billion curies, for a specific activity of 100,000 Ci per MTHM. For this EIS, it is assumed that one Pegase cask contains 0.0155 MTHM (15.5 kg) of spent nuclear fuel and 930,000 Ci, for a specific activity of 60 million Ci per MTHM. Table C-14 contains a detailed list of the inventory of radionuclides for both the Nuclear Energy Agency vitrified high-level waste and the foreign research reactor spent nuclear fuel. The specific activity for the vitrified high-level waste is significantly lower than that of the foreign research reactor spent nuclear fuel because the Nuclear Energy Agency study uses data assuming a 100-year decay time for the waste, while the foreign research reactor spent nuclear fuel is assumed to only have been out of the reactor less than a year. The Nuclear Energy Agency study used 100-year decay time because in their study the spent nuclear fuel was not vitrified until it was 50 years out of the reactor, and it was assumed to take 50 years for their cask to fail once it was in the ocean.

The dose estimates from the Nuclear Energy Agency analysis are scaled for this EIS to reflect (1) the fact that spent nuclear fuel corrodes faster than vitrified glass, (2) there is significantly less mass of heavy metal in a spent nuclear fuel cask than was used in the Nuclear Energy Agency dose risk models, and (3) the specific activity of the foreign research reactor spent nuclear fuel is higher than the specific activity of the Nuclear Energy Agency vitrified high-level waste.

To account for differences in the waste matrix corrosion rate, the sensitivity of the calculated dose to the corrosion rate was used. In its radiological assessment, the Nuclear Energy Agency published sensitivity studies. For the accident analyses, an adjoin method was used to determine the sensitivity of the peak individual dose and the collective dose to key parameters in their performance assessment model, including the waste matrix corrosion rate.

The adjoin method employs a mathematical algorithm for calculating directly in one run the sensitivity of a performance assessment model to the model parameters. It gives as output the first derivative of the response of the performance assessment model (here, peak individual dose and collective dose) with respect to each of the model parameters (in particular, corrosion rate). Explicitly, the sensitivity coefficient is defined as:

$$a = \frac{\partial D/D}{\partial \alpha/\alpha} \qquad (1)$$

where a is the sensitivity coefficient, D is the dose (peak or cumulative), and α is a given parameter (leach rate). This expression can be used to determine the change in the dose for a change in the parameter value by integrating as follows in equation 2.

Total

$$\int_{D_0}^{D_1} \frac{\partial D}{D} = a \int_{\alpha_0}^{\alpha_1} \frac{\partial \alpha}{\alpha} \qquad (2)$$

or

Table C-14 Comparison of Radionuclide Inventories for Nuclear Energy Agency High-Level Waste Sub-Seabed Disposal Studies and BR-2 Foreign Research Reactor Spent Nuclear Fuel

Radionuclide	*Nuclear Energy Agency Vitrified High-Level Waste Inventory[a] (Ci)*	*Foreign Research Reactor Spent Nuclear Fuel Inventory[b] (Ci)*	*Radionuclide*	*Nuclear Energy Agency Vitrified High-Level Waste Inventory[a] (Ci)*	*Foreign Research Reactor Spent Nuclear Fuel Inventory[b] (Ci)*
Hydrogen-3	0.0	86.4	Cerium-141	0.0	5,700
Selenium-79	33,000	0.0	Cerium-144	0.0	310,000
Krypton-85	0.0	2,500	Promethium-147	11,000	48,000
Strontium-89	0.0	41,000	Promethium-148m	0.0	75.6
Strontium-90	2,000,000,000	21,000	Samarium-151	27,000,000	0.0
Yttrium-90	2,000,000,000	0.0	Europium-154	8,600,000	620
Yttrium-91	0.0	73,000	Europium-155	480,000	130
Niobium-95	0.0	220,000	Uranium-233	178	0.0
Zirconium-93	180,000	0.0	Uranium-234	300	0.0091
Zirconium-95	0.0	110,000	Uranium-235	0.0	0.014
Technicium-99	1,400,000	0.0	Uranium-236	47	0.0
Ruthenium-103	0.0	8,900	Uranium-238	0.0	0.00034
Ruthenium-106	0.0	22,000	Neptunium-237	32,000	0.0
Palladium-107	10,000	0.0	Plutonium-238	0.0	64.2
Tin-123	0.0	430	Plutonium-239	120,000	1.8
Tin-126	58,000	0.0	Plutonium-240	620,000	1.2
Antimony-125	990	890	Plutonium-241	3,500,000	280
Antimony-126m	58,000	0.0	Plutonium-242	600	0.0
Tellurium-125m	0.0	210	Americium-241	6,900,000	0.4
Tellurium-127m	0.0	890	Americium-242m	0.0	0.0011
Tellurium-129m	0.0	200	Americium-243	2,000,000	0.0043
Iodine-129	3.0	0.0	Curium-242	0.0	1.8
Cesium-134	108	16,000	Curium-244	0.0	1.3
Cesium-135	150,000	0.0	Curium-245	21,000	0.0
Cesium-137	3,000,000,000	21,000	Curium-246	5,500	0.0
Barium-137m	2,900,000,000	0.0			
			Total	**10,000,000,000**	**930,000**

[a] *Nuclear Energy Agency vitrified high-level waste radionuclide inventories are based on 100,000 MTHM that represent spent nuclear fuel radionuclide inventories for 100 years out of reactor. The Nuclear Energy Agency analysis based its dose rate estimate calculations on vitrified high-level waste that was produced from commercial light water reactor spent nuclear fuel at 50 years out of reactor, then the Nuclear Energy Agency analysis models the release of the vitrified high-level waste inventory into the ocean only after an additional 50 years of submersion.*

[b] *Foreign research reactor spent nuclear fuel radionuclide inventories are based on a Pegase cask filled with 36 elements of BR-2 spent nuclear fuel, 300 days out of reactor.*

$$\ln (D_1/D_0) = a \ \ln (\alpha_1/\alpha_0) \tag{3}$$

Using the data provided in Table C-13,

$$\ln (D_1/D_0) = 0.99 \ln (8.6 \, x \, 10^{-3}/3.6 \, x \, 10^{-5}) \tag{4}$$

or

$$D_1 = 227 \, D_0 \tag{5}$$

Where D_1 is the dose by foreign research reactor spent nuclear fuel, adjusted only for the difference in leach rate, and D_0 is the Nuclear Energy Agency dose.

Since the derivative in Equation (1) is evaluated at a particular value of each model parameter, it is by definition the sensitivity coefficient of the dose to small variations in each parameter around their assigned value. As a result, the calculation of dose using the sensitivity coefficient is valid only when changes in the leach rate remain "sufficiently small" compared to the leach rate. However, the Nuclear Energy Agency assessment states that many of the models in their assessment are linear, and it is possible to estimate changes in the dose even for large variations in the leach rate.

To account for differences in the waste inventory, the dose was scaled linearly according to the ratio of the specific activity of the BR-2 spent nuclear fuel to the specific activity of the vitrified high-level waste as shown in Equation (6).

$$D = D_1 \frac{\beta_{EIS}}{\beta_{NEA}} = D_1 \frac{0.0155}{1.0E{+}05} \frac{6.0E{+}07}{1.0E{+}05} = 9.3E{-}05 \ D_1 \tag{6}$$

Finally,

$$D = 0.021 \ D_0 \tag{7}$$

C.5.4 Results

Dose rates were calculated in the Nuclear Energy Agency study for two types of ocean environments, coastal waters and deep ocean floors. The results of scaling the Nuclear Energy Agency dose rate estimates for the scenario of losing a cask of foreign research reactor spent nuclear fuel in coastal waters are shown in Table C-15, with the comparable Nuclear Energy Agency results. In Table C-16, the results of losing a cask containing foreign research reactor spent nuclear fuel in deep ocean waters are shown. Table C-15 presents results for both an undamaged and a damaged cask, however Table C-16 provides the estimated dose for a damaged cask only because it is assumed that the pressure from the deep ocean will damage the cask seals.

The doses associated with the foreign research reactor spent nuclear fuel in Table C-16 are, in the case of the mollusks, very high. However, to properly interpret this result, several factors must be considered. First, the calculation that produced these results is very conservative for two reasons. The radioactive material, once corroded, was assumed to immediately be released into the open ocean water. In fact, the cask is expected to provide a significant "hold-up" time. This is because only the seal is expected to fail, which means that, due to the small area of the seal, only a very limited amount of water movement through the cask will be experienced. Over time, this small flow would carry out all of the soluble fission products, but insoluble precipitates would remain in the cask. Also, no account was taken for the possibility that the cask would likely become buried in silt, greatly slowing the fission product's entry into

Table C-15 Coastal Waters Dose Rate Estimates for 100,000 MTHM Vitrified High-Level Waste and a Pegase Cask Loaded With BR-2 Foreign Research Reactor Spent Nuclear Fuel

Dose Category	*D_0 (NEA)*	*D (BR-2)*
Undamaged Cask Peak Individual Dose	9.0 rem/yr	0.19 rem/yr
Damaged Cask Peak Individual Dose	650 rem/yr	14 rem/yr
Undamaged Cask Peak Biota Dose (Fish)	3.6 rad/yr	0.077 rad/yr
Undamaged Cask Peak Biota Dose (Crustaceans)	3.8 rad/yr	0.081 rad/yr
Undamaged Cask Peak Biota Dose (Mollusks)	10 rad/yr	0.21 rad/yr
Damaged Cask Peak Biota Dose (Fish)	29 rad/yr	0.62 rad/yr
Damaged Cask Peak Biota Dose (Crustaceans)	31 rad/yr	0.66 rad/yr
Damaged Cask Peak Biota Dose (Mollusks)	660 rad/yr	14 rad/yr

Table C-16 Deep Ocean Dose Rate Estimates for 100,000 MTHM Vitrified High-Level Waste and a Pegase Cask Loaded with BR-2 Foreign Research Reactor Spent Nuclear Fuel

Dose Category	*D_0 (NEA)*	*D (BR-2)*
Damaged Cask Peak Individual Dose	0.00053 rem/yr	0.114 rem/yr
Damaged Cask Peak Biota Dose (Fish)	30,000 rad/yr	640 rad/yr
Damaged Cask Peak Biota Dose (Crustaceans)	41,000 rad/yr	880 rad/yr
Damaged Cask Peak Biota Dose (Mollusks)	1,400,000 rad/yr	30,000 rad/yr

the open water. Also, no account was taken of the reduction in corrosion rate in the deep ocean due to lower oxygen levels or the reduced temperatures. These factors indicate that if a rigorous calculation were possible, the resultant dose would be lower, and likely significantly lower.

Once out of the cask, the fission products are unlikely to be transported very far in the very slow current typical in the deep ocean. While this would concentrate the dose to those organisms in the area of the cask, especially the mollusks, it also means that the population affected would be relatively small, since only a small area would be contaminated.

Additionally, as explained in Chapter 3, the density of organisms in the deep ocean is around one percent that in coastal waters. This further reduces the affected population of organisms.

The risks associated with the dose estimated for the mollusk are very low, due to the low frequency of the event, as explained in the following section.

C.5.5 Risks Associated With Submersion of a Foreign Research Reactor Spent Nuclear Fuel Cask

Risks associated with submersion of foreign research reactor spent nuclear fuel casks were calculated for a single cask, even though more than one cask may be carried on some voyages. The risk (consequences multiplied by probability) is essentially independent of the number of casks carried per voyages. That is, the risk associated with eight voyages of one cask each are essentially the same as one voyage carrying eight casks.

The consequence estimates in Tables C-15 and C-16 are indicative of what could happen in the event that a foreign research reactor spent nuclear fuel cask were to become submerged in coastal waters or in the deep ocean and is not recovered. By combining an estimate of the frequency at which such a situation is expected to occur with the consequence estimates, an estimate of the risk associated with ocean transportation can be developed. The frequency of a cask becoming submerged is: the mathematical product of the annual frequency of foreign research reactor spent nuclear fuel shipments, the probability that a shipment is involved in an accident, the probability that a ship sinks (given that an accident occurs), and the probability that a submerged cask is not recovered. Additionally, the frequency of a damaged cask becoming submerged in coastal waters includes the probability that a cask is damaged given that an accident occurs. The data for these events were taken from two sources, the Nuclear Energy Agency study (NEA, 1988) and the Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994). These data are summarized in Table C-17.

Table C-17 At Sea Risk Assessment Data

Parameter	*Coastal Data*	*Deep Ocean Data*
Shipment Accident Rate	0.00032/Shipment (DOE, 1994)	0.000046/Shipment (NEA, 1988)
Probability that Cask is Damaged, Given an Accident	0.002 (DOE, 1994)[a]	1.0[c]
Probability that a Ship Sinks Given an Accident	0.001 (Wheeler, 1994)	0.001 (Wheeler, 1994)
Probability that a Submerged Cask is not Recovered	0.0001 (NEA, 1988)[b]	0.05 (NEA, 1988)
Number of Shipments	721	721
Probability - Submerged Cask, Damaged, Unrecovered	4.6×10^{-11}	0.0000017
Probability - Submerged Cask, Undamaged, Unrecovered	2.3×10^{-8}	0.0c

[a] *This value represents the conditional probability that the severity of an accident is greater than Category II, as shown in Appendix E, Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994).*

[b] *Derivation of this probability is based in a fault tree analysis using data from the Nuclear Energy Agency analysis.*

[c] *The cask is assumed to fail at deep ocean depths.*

The risk estimate results for the basic implementation of Management Alternative 1 are shown in Table C-18. The risk for a peak dose to an individual is 6.4×10^{-7} mrem per year for a damaged cask in coastal water and 0.0000043 mrem per year for an undamaged cask. Risk associated with a submerged, unrecovered cask in the deep ocean is 0.00019 mrem per year for a damaged cask.

Table C-18 Radiological Risk Estimates for At Sea Accidents

Dose Category	*Damaged Cask*	*Undamaged Cask*
Coastal Dose Rate Risk Estimates		
Peak Individual Dose	6.4×10^{-7} mrem/yr	0.0000043 mrem/yr
Peak Biota Dose (Fish)	2.8×10^{-8} mrad/yr	0.0000018 mrad/yr
Peak Biota Dose (Crustaceans)	3.0×10^{-8} mrad/yr	0.0000019 mrad/yr
Peak Biota Dose (Mollusks)	6.4×10^{-7} mrad/yr	0.0000048 mrad/yr
Deep Ocean Risk Estimates		
Peak Individual Dose	0.00019 mrem/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Fish)	1.1 mrad/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Crustaceans)	1.4 mrad/yr	Cask is assumed to fail at deep ocean depths
Peak Biota Dose (Mollusks)	49 mrad/yr	Cask is assumed to fail at deep ocean depths

C.5.6 Marine Accident Impacts of Policy Alternatives

In Section C.4.2, two implementation subalternatives to Management Alternative 1 and one implementation subalternative to Management Alternative 2 of the proposed action that could impact the risk calculations were identified: accepting spent nuclear fuel from developing countries only, a 5-year acceptance program, and overseas reprocessing of the foreign research reactor spent nuclear fuel. Implementation of any of these has the potential to impact the marine accident risks calculated for the basic implementation of Management Alternative 1 calculated above.

For the implementation subalternatives involving the shipment of different quantities of foreign research reactor spent nuclear fuel, the consequences of an accident are the same for the implementation subalternatives as they are for the basic implementation. In these subalternatives, the same type of spent nuclear fuel is being shipped in the same types of transportation casks and is subject to the same accidents as for the basic implementation. These are the variables between subalternatives that could have affected the consequences of a marine accident. Since none changed, the consequences do not change. Two of the implementation subalternatives fall into this category: the developing countries implementation subalternative and the five-year policy duration implementation subalternative. For these two alternatives, the marine accident risks are directly proportional to the number of foreign research reactor spent nuclear fuel shipments required to implement each implementation subalternative. It is therefore possible to scale the results presented in the previous section by the ratio of the number of cask shipments in implementation subalternative to the number of cask shipments in the basic implementation.

Subalternative 1b to Management Alternative 2 requires the shipment of the foreign research reactor spent nuclear fuel wastes in a different form than the basic implementation. With overseas reprocessing of the foreign research reactor spent nuclear fuel, any material that would be returned to the United States would be in the form of vitrified high-level waste. As discussed earlier in Section C.5, the high-level waste behaves differently when exposed to seawater than does spent nuclear fuel. The vitrified waste dissolves at a much slower rate than the foreign research reactor spent nuclear fuel. A second major difference is the amount of radioactivity present in each of the shipping casks carrying vitrified waste and spent nuclear fuel. As shown in Table C-14, the total curie content of a transportation cask carrying foreign research reactor spent nuclear fuel is approximately a million curies. Each vitrified waste transportation cask could contain approximately a hundred times this amount. The contents of 837 spent nuclear fuel transportation casks (all foreign research reactor spent nuclear fuel could be processed, including that from Canada, which was not included in the marine risk analyses for the basic implementation) are expected to be reduced to fit into eight transportation casks.

In addition, a Hybrid Alternative has been analyzed to assess the impact of encouraging overseas reprocessing of the foreign research reactor spent nuclear fuel for those countries capable of storing the resultant high-level waste. The United States would accept for management the research reactor spent fuel from countries that are unable to accept and store the high-level waste resulting from fuel processing. Under the Hybrid Alternative analyzed, Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom are assumed to process their aluminum-based spent nuclear fuel and accept the high-level waste. All other countries identified in Table C-1 would be allowed to ship spent nuclear fuel to the United States. The resulting 452 shipments of spent nuclear fuel (excluding the overland transport of fuel of Canadian origin) are the basis for the marine impact analysis for this Hybrid Alternative.

The marine accident risks associated with each of these management alternatives is presented in the following paragraphs.

Management Alternative 1, Implementation Subalternative 1a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only from Developing Countries: This implementation subalternative would result in the shipment of 168 transportation casks of foreign research reactor spent nuclear fuel. This is 23 percent of the shipments required for the basic implementation. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The maximally exposed individual (MEI) would be exposed to a risk (in terms of a peak individual dose rate) of 0.000044 mrem per year as a result of an accident causing the loss of a cask in the deep ocean. The consequences of this accident do not change; the peak individual dose remains at 0.114 rem per year. The loss of a damaged cask in coastal waters results in the lowest risk to man, 1.5 x 10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 11 mrad per year to a mollusk from the loss of a cask in the deep ocean, to a low of 6 x 10^{-9} mrad per year to fish from the loss of a damaged cask in coastal waters.

Management Alternative 1, Implementation Subalternative 2a — Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5-Year Policy Duration: This implementation subalternative results in the shipment of 586 transportation casks of foreign research reactor spent nuclear fuel. This is 81 percent of the shipments required for the basic implementation. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The MEI will be exposed to a risk (in terms of a peak individual dose rate) of 0.00015 mrem per year as a result of the accident causing the loss of a cask in the deep ocean. The loss of a damaged cask in coastal waters results in the lowest risk to man, 5 x 10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 40 mrad per year to a mollusk (deep sea accident) to a low of 2 x 10^{-8} mrad per year to fish (coastal water, damaged cask accident).

Management Alternative 2, Subalternative 1b — Overseas Processing with Shipment of Waste to a U.S. Storage Facility: In this subalternative, all of the foreign research reactor spent nuclear fuel (including that generated in Canada) is sent to either Great Britain or France for processing and the vitrified high-level waste generated in the process would be shipped to the United States. Based on the processing of approximately 23 metric tons (25.3 tons) of spent nuclear fuel, enough vitrified high-level waste would be generated to require up to eight transportation casks of vitrified high-level waste being shipped to the United States. Only the impact of the marine shipments from the processing facility to the United States was calculated.

The consequences of an accident at sea that results in the loss of a transportation cask filled with vitrified high-level waste can be derived from the information used to develop the marine accident consequences for a foreign research reactor spent nuclear fuel cask. The consequences listed in Tables C-15 and C-16 for D_0 represent the consequences associated with the loss of 100,000 MTHM equivalent of vitrified high-level waste. Based on eight shipments for the approximately 23 metric tons (25.3 tons) of spent nuclear fuel, each shipment in this subalternative will contain approximately 2.9 metric tons (3.2 tons) equivalent of vitrified high-level waste. Table C-19 presents the consequences from Tables C-15 and C-16 scaled to represent the consequences for an accident resulting in the loss of a transportation cask containing 2.9 metric tons (3.2 tons) equivalent.

Table C-19 Consequences Resulting from the Loss of a Transportation Cask Containing Vitrified High-Level Waste[a]

	Coastal Waters		*Deep Ocean*
	Undamaged Cask	*Damaged Cask*	*Damaged Cask*
Peak Individual Dose (Man) rem/yr	0.0003	0.019	1.5 x 10^{-8}
Peak Biota Dose (Fish) rad/yr	0.0001	0.0008	0.9
Peak Biota Dose (Crustaceans) rad/yr	0.0001	0.0009	1.2
Peak Biota Dose (Mollusks) rad/yr	0.0003	0.019	41

[a] *These estimates are based on the best estimate values presented in the Nuclear Energy Agency report (NEA, 1988)*

From the accident frequency data in Table C-17, a per-shipment accident frequency can be developed for all three accidents of interest: 1) the loss of an undamaged cask in coastal waters, 2) the loss of a damaged cask in coastal waters, and 3) the loss of a damaged cask in the deep ocean. These frequencies are the product of the shipment accident rate, the probability of the vessel sinking after an accident, the probability that a submerged cask is not recovered, and where applicable (for the damaged cask in coastal waters only), the probability that the cask is damaged in the accident. The resulting per shipment accident probabilities are 3.2×10^{-11} for the loss of an unrecovered, undamaged cask in coastal waters, 6.4×10^{-14} for the loss of an unrecovered damaged cask in coastal waters, and 2.3×10^{-9} for the unrecovered loss of a damaged cask in the deep ocean.

With the assumption that there are only up to eight shipments of vitrified high-level waste, the risks associated with the marine transport of this material are almost non-existent. The risks in terms of rem per year peak public dose and rad per year peak dose to marine biota, of an unrecovered cask in coastal waters are essentially zero, less than 1.0×10^{-10}. The risks calculated for the deep ocean accidents are: much less than 1×10^{-10} rem per year peak dose to man, 2×10^{-8} rad per year peak dose to fish and crustaceans, and 7×10^{-7} rad per year peak dose to mollusks.

Management Alternative 3 — Combination of Components of Management Alternatives 1 and 3 (Hybrid Alternative): Under the Hybrid Alternative, the United States would accept foreign research reactor spent nuclear fuel from countries unable to store high-level waste. This Hybrid Alternative could result in the shipment of 452 transportation casks of foreign research reactor spent nuclear fuel to the United States. This is approximately 63 percent of the shipments required in the basic alternative. Using this relationship, the risks presented in Table C-18 can be scaled to produce the following results. The MEI will be exposed to a risk (in terms of a peak individual dose rate) of 0.00012 mrem per year as a result of an accident causing the loss of a cask in the deep ocean. The consequences of this accident do not change from the basic implementation; the peak individual dose remains at 0.114 mrem per year. The loss of a damaged cask in coastal waters results in the lowest risks to man, 4×10^{-7} mrem per year. The risks to marine biota are reduced by the same ratio and will range from a high of 31 mrad per year to a mollusk (deep sea accident) to a low of 1.8×10^{-8} mrad per year to fish (coastal water, damaged cask accident).

References

DMA (Defense Mapping Agency), 1991, "Distances Between Ports," Publication 151, No. NVPUB151, DMA, Washington DC.

DOC (U.S. Department of Commerce), 1994, *Statistical Abstract of the United States, 114th Edition, Bureau of Census,* Washington, DC.

DOE (U.S. Department of Energy), 1994, *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel,* DOE/EA-0912, Washington, DC, April.

Encyclopedia Americana, 1991, Volume 28.

IAEA (International Atomic Energy Agency), 1990, *Regulations for the Safe Transport of Radioactive Material,* Safety Series No. 6, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection,* Annals of the ICRP, Publication 60, Volume 21, No. 1-3, Pergamon Press, New York, November.

NEA (Nuclear Energy Agency), 1988, *Feasibility of Disposal of High-Level Radioactive Waste into the Seabed, Volume 2: Radiological Assessment,* OECD, Paris.

Rechard, R., 1994, *Initial Performance Assessment of the Disposal of Spent Nuclear Fuel and High-Level Waste Stored at Idaho National Engineering Laboratory,* SAND93-2330, Sandia National Laboratories, Albuquerque, NM.

Wheeler, T.A., 1994, Sandia National Laboratories, personal communication with K. Neuhauser, October.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix D
Selection and Evaluation of Potential Ports of Entry

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

List of Tables

Appendix D
Selection and Evaluation of Potential Ports of Entry

This appendix describes the process used by the Department of Energy (DOE) in selecting the potential ports of entry analyzed in this Environmental Impact Statement (EIS). In addition, the appendix provides the basic information required to evaluate ports and port activities, and the potential environmental impacts (incident-free and accidents) associated with the receipt and handling of foreign research reactor spent nuclear fuel from vessels to intermodal transport in ports.

D.1 Ports of Entry Selection Process

The adopted port selection process was based on a set of criteria developed by DOE to identify those ports that would be most capable of providing for the safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. This appendix first describes the process through which DOE developed the port selection criteria, and then describes the application of the criteria, resulting in the identification of the specific ports available for consideration.

Because the basic implementation of Management Alternative 1 of the proposed action would involve shipments from many foreign countries to several potential foreign research reactor spent nuclear fuel management sites in addition to those identified in the Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d), it was necessary to expand on the port analysis approach used in the Urgent Relief Environmental Assessment. The Urgent Relief Environmental Assessment was concerned with fewer shipments within a short timeframe, with the shipments going only to the Savannah River Site. Also, as stated in the Urgent Relief Environmental Assessment, this EIS considers future population trends and future port developments.

Independent maritime safety experts consulted during the preparation of this EIS informed DOE that any modern breakbulk or container terminal can accommodate the safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel in approved shipping casks (USMMA, 1994). This view is supported by the well-documented excellent safety record for shipping foreign research reactor spent nuclear fuel in the United States (NRC, 1993). In addition, the port selection criteria discussed in this appendix, taken collectively, provide a reasonable additional basis for identifying those candidate ports suitable for the safe receipt and handling of foreign research reactor spent nuclear fuel.

D.1.1 Background

Since 1979, when the U.S. Nuclear Regulatory Commission (NRC) first began approving spent nuclear fuel shipments in the United States, 317 spent nuclear fuel shipments in Type B casks have been transported safely into several U.S. ports of entry. These include Newport News, Norfolk, and Portsmouth, VA; Portland, OR; Savannah, GA; and Oakland, CA (NRC, 1993). However, prior to the fall of 1993, DOE did not have any generally applicable criteria for identifying ports of entry. For this EIS, as well as for the Urgent Relief Environmental Assessment, DOE developed criteria to identify candidate ports of entry. The criteria used in this evaluation to identify potential ports for the receipt of foreign research reactor spent nuclear fuel are based on consideration of several independent factors, each described in more detail in the following sections.

D.1.2 Information on Past Spent Nuclear Fuel Shipments

The NRC has the authority under the Atomic Energy Act of 1954, as amended, to regulate certain aspects of spent nuclear fuel transportation. Of the thousands of shipments completed over the last 30 years in the United States and abroad, none has resulted in an injury due to the radioactive nature of the cargo (NRC, 1993). For the same period, about 1,200 (924 domestic and 293 foreign) overland shipments of spent nuclear fuel took place without any injury attributable to accidents or incident-free radiation doses during transport. Table D-1 provides the number of NRC and Department of Transportation regulated international and domestic overland shipments since 1979 (excluding DOE shipments). The casks that would be used in this program are robust Type B containers. The safety, safeguards, and precautions used for such shipments have historically been very successful (NRC, 1993).

Table D-1 Number of NRC/Department of Transportation Regulated Overland and International Spent Nuclear Fuel Shipments

Year	Overland		International[a]		
	Highway	Railway	Export	Import	Transient
1979	2	11	0	14	0
1980	73	5	2	55	0
1981	30	2	3	48	0
1982	80	0	1	43	0
1983	92	0	2	23	0
1984	209	3	2	34	0
1985	114	18	0	21	0
1986	88	15	0	17	0
1987	85	15	3	19	0
1988	10	7	0	15	0
1989	11	6	1	4	0
1990	0	8	2	0	3
1991	7	10	4	0	1
1992	17	6	0	0	0
Total	**818**	**106**	**20**	**293**	**4**

[a] *Ports included Newport News, Norfolk, and Portsmouth, VA; Savannah, GA; Portland, OR; and Oakland, CA.*

D.1.3 Federal Court Ruling

In the December 1991 decision of the U.S. District Court, District of Columbia Circuit, on the return of spent nuclear fuel from Taiwan, the court ruled that DOE must consider a reasonable range of alternative ports, including (at least) two low population density ports near DOE's Savannah River Site (U.S. District Court, 1991). In this appendix, DOE has identified a reasonable range of alternative ports on the East, Gulf, and West Coasts (including several low population ports) for the receipt, handling, and transhipment of foreign research reactor spent nuclear fuel to five potential DOE management sites (including the Savannah River Site) being considered in this EIS.

D.1.4 Notice of Intent Port Criteria

The Notice of Intent for this EIS (DOE, 1993) listed a series of preliminary criteria which might be applied to a potential list of candidate seaports to identify ports which would be acceptable for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. These proposed criteria included: "(a) adequacy of harbor and dock characteristics to satisfy the cask carrying ship requirements;

(b) availability of safe and secure lag storage; (c) adequacy of overland transportation systems from ports to the storage site(s); (d) experience in safe and secure handling of hazardous cargo; (e) emergency preparedness status at the port and nearby communities; and (f) proximity to the proposed storage sites." Either implicitly or explicitly, these criteria were considered in the port screening, as discussed in the following sections.

D.1.5 The U.S. Merchant Marine Academy Workshop Recommendations

A DOE-sponsored workshop on port selection criteria for spent nuclear fuel was held at the U.S. Merchant Marine Academy at Kings Point, New York, on November 15-16, 1993 (USMMA, 1994). Participants at the workshop included experts from the maritime industry in the areas of marine transportation, intermodal systems, marine insurance, admiralty law, U.S. Coast Guard Operations, U.S. Navy Operations, Military Sealift Command Operations, and national cargo, pilotage, and ships operations.

A series of panel discussions focused on issues such as economics and transportation safety, advantages of shipping spent nuclear fuel on various types of vessels, and shipping spent nuclear fuel through large versus small ports. The purpose of such discussions, in part, was to enable DOE to identify port criteria that would minimize both the actual and perceived risk involved in spent nuclear fuel shipments. The workshop participants agreed that any port capable of handling an ocean-going vessel is capable of receiving spent nuclear fuel. While some of these ports might have features that would make them more desirable than others (e.g., easy access from the open sea, modern facilities, etc.), no port would have such limitations as to preclude safe receipt of the spent nuclear fuel. While individual ports might not satisfy all the criteria recommended at the workshop, the workshop participants concluded that the criteria would provide a means of evaluating the relative merits of ports.

The three criteria recommended as necessary for safe shipment were: short distance from the open ocean to the port, adequate port cargo facilities, and intermodal access (i.e., for truck or rail shipments from the port to the management site).

A second set of recommendations that were listed as "important but not necessary" included: an experienced risk management staff, emergency preparedness and response capabilities, a skilled labor force aboard ship and in port, good port security, no local restrictions or regulations on movement of hazardous cargo, and no significant environmental considerations for the port.

Finally, the workshop also provided a list of "desirable" attributes for ports, including: distance of the port from a population center, proximity of the port to a spent nuclear fuel management location, "local economic issues" (e.g., areas that receive a significant fraction of their revenues from maritime and shipping activities), and personnel with training and experience in radioactive shipments and incident response.

D.1.6 Provisions of the National Defense Authorization Act for Fiscal Year 1994

On November 30, 1993, the National Defense Authorization Act for Fiscal Year 1994 was signed into law (NDAA, 1993). Section 3151 stipulates specific criteria that must be used "if economically feasible" and "to the maximum extent practicable" in selecting U.S. ports for both emergency and nonemergency receipt of foreign research reactor spent nuclear fuel at the Savannah River Site. Although the National Defense Authorization Act does not specifically address other potential DOE management sites, DOE assumed that the guidance provided for foreign research reactor spent nuclear fuel shipments to Savannah

River Site should be considered for the other four potential sites being considered in this EIS (Idaho National Engineering Laboratory, Hanford Site, Oak Ridge Reservation, and the Nevada Test Site), to the extent feasibility and practicability permitted.

Specifically, the National Defense Authorization Act requires that DOE may not receive foreign research reactor spent nuclear fuel if it "cannot be transferred in an expeditious manner from its port of entry in the United States to a storage facility that is located at a Department of Energy facility and is capable of receiving and storing the spent nuclear fuel." Further, it requires that the "Secretary of Energy shall, if economically feasible and to the maximum extent practicable, provide for the receipt of spent nuclear fuel....at a port of entry in the United States which...compared to each other port of entry....that is capable of receiving the spent nuclear fuel - (1) has the lowest human population in the area surrounding the port of entry; (2) is closest in proximity to the facility which will store the spent nuclear fuel; and (3) has the most appropriate facilities for, and experience in, receiving nuclear fuel (NDAA, 1993)."

D.1.7 Comments Received During the EIS Scoping Meetings and on the Urgent Relief Environmental Assessment

Nine public scoping meetings were held in November and December, 1993, at six cities being considered as potential ports for the receipt of spent nuclear fuel from foreign research reactors, and four cities near the potential spent nuclear fuel management sites discussed in this EIS. As a result of these meetings, DOE received several groups of similar comments, which have been incorporated into the development of the criteria (DOE, 1994a).

The largest number of comments (44) received on any general port-related issue dealt with avoiding ports in high population areas. Reasons ranged from concerns about accident consequences and possible terrorist attacks, to concerns about the ability to adequately respond to emergencies and possible evacuation of populations.

The second largest number of comments (32) suggested that alternative ports in low-population areas or ports operated by the military be seriously considered, and that ports that are closest to the storage sites and/or have the most direct transportation routes between the ports and management sites be considered.

Other comments that fall within the jurisdiction of DOE and within the scope of this EIS include: suggestions that selected ports should have experience handling spent nuclear fuel (9 comments); the safest marine terminals should be used at the port selected (3 comments); and that DOE should allow case-by-case designation of ports based on the most sensible options at the time each individual shipment occurs, considering the vessel, country of origin, time, cost, and overall experience of the ports (2 comments).

In addition to comments presented at the EIS Scoping Meetings, DOE has also considered individual comments and a list of suggestions from the Sierra Club on the draft Urgent Relief Environmental Assessment (DOE, 1994d).

D.1.8 Key Assumptions and Methodology for Port Identification

A number of possible maritime shipment modes are potentially available for shipping the foreign research reactor spent nuclear fuel over the next 10 or more years. The various transport modes generally determine which port facilities are adequate at each specific port [e.g., container cranes are required for container vessels, a pier for roll-on/roll-off vessels, and breakbulk cranes for breakbulk vessels]. While regularly scheduled cargo ships servicing commercial ports could be an important mode selected by owners of the foreign research reactor spent nuclear fuel for their shipments, smaller unscheduled vessels would also be a common mode of transport for multiple cask shipments (e.g., the first shipment of foreign research reactor spent nuclear fuel under the Urgent Relief Environmental Assessment in September 1994). This means that there will be a somewhat greater number of potential ports of entry to consider than if only larger, regularly scheduled commercial container vessels were to be used (details on potential vessel types that might be used are provided in Appendix C).

In addition to the types of vessels that could be used, the way foreign research reactor spent nuclear fuel casks are "packaged" for shipment is also a determinant in the selection of potential ports. For the Urgent Relief Environmental Assessment shipments to Savannah River Site, the Terms and Conditions for Financial Settlement for Receipt and Disposition of Foreign Research Reactor Spent Fuels (DOE, 1994c) required that spent nuclear fuel casks be containerized in 20 ft International Standards Organization containers (nominally, 2.4 m x 2.4 m x 6.1 m, or 8 ft x 8 ft x 20 ft), also called 20-ft equivalent units. Therefore, it was assumed that spent nuclear fuel casks would only be shipped containerized. This eliminates consideration of receipt and handling of foreign research reactor spent nuclear fuel casks in a "palletized" mode. Thus, the EIS focuses primarily on reasonable options for ports qualified for the receipt, handling, and transhipment of containerized spent nuclear fuel on any viable vessel type.

Among the ports that routinely handle containerized freight, two groups of ports - those along the in-land Mississippi River (above New Orleans) and those around the Great Lakes - are not considered in this evaluation. Access to these ports requires a long inland transit from open ocean. The U.S. Merchant Marine Academy recommendations discouraged such transits.

Finally, since the National Defense Authorization Act did not establish numerical distance or transport time limits for spent nuclear fuel transport, DOE concluded that, consistent with current and past Federal practice for transport of spent nuclear fuel in the contiguous United States, all overland shipments should be managed such that the spent nuclear fuel is kept moving as expeditiously as possible from the time it is placed on the transportation vehicle at the port of entry until it reaches the DOE management site, to the maximum extent practicable. For example, truck shipments (which typically involve two drivers in a tractor with a sleeping area) are assumed to be basically nonstop in order to deliver the spent nuclear fuel promptly, stopping only for fuel and food. This has been, and is expected to remain, DOE practice for such shipments.

NRC recently reported that for the period 1979-1992, rail transport only accounted for 8.6 percent of the total spent nuclear fuel shipments in the United States, but these shipments accounted for about 66 percent of the total quantity of spent nuclear fuel shipped (NRC, 1993). Rail travel (freight) typically takes much longer than truck transport when moving spent nuclear fuel on a dedicated railcar, where even short trips may require movement through additional intermodal terminals (e.g., transfer from rail to truck for site delivery), or intermediate points of dedicated railcar transfers to other train systems (e.g., from a local freight handler to one or more long distance freight lines). However, in the case of dedicated trains where entire multiple cask shipments (such as those used for the Urgent Relief shipment from the Military Ocean Terminal at Sunny Point (MOTSU) to the Savannah River Site) go directly from the port to the management site, rail travel times are expected to be somewhat longer than those for truck transport. Generally, rail distances are also typically somewhat longer than those for trucks using interstates, and rail transport generally costs more, and potentially exposes larger numbers of people since transits typically pass through major railyards in inner cities (see Appendix E for comparative travel distances for truck and rail).

In both cases, DOE concludes that by proper planning and compliance with current Department of Transportation and NRC shipment requirements (including use of pre-approved routes), each shipment of foreign research reactor spent nuclear fuel could be moved expeditiously from each port to each management site, and specific distance and time considerations do not serve to usefully discriminate against ports in the contiguous 48 States.

D.1.9 Methodology for Port Selection

The methodology for identifying acceptable ports of entry began with a list of 153 commercial ports throughout the contiguous United States. These ports included the 151 ports that were originally considered in the Urgent Relief Environmental Assessment (DOE, 1994d). The two additional commercial ports are Eddystone, PA, and Fernandina Beach, FL. Also, eight additional military ports in the contiguous United States suggested by the Military Traffic Management Command (MTMC, 1994a) were evaluated. The eight candidate military ports were those believed to routinely handle dry containerized cargoes (largely munitions), on breakbulk, container, and/or roll-on/roll-off vessels. Military ports are subject to extreme fluctuations in port activities as a function of national need. By using the criteria described below, ports that did not meet each DOE mandatory criterion in the sequence were eliminated. Those ports not eliminated at each step of the screening process were then evaluated in the same fashion against the remaining required criteria.

The required screening criteria DOE used to identify potential ports of entry are:

- The ports must have appropriate (routine) experience handling containerized cargo (Criterion 1);
- The ports must offer favorable transits from the open ocean to the selected terminals (Criterion 2);
- The ports must have appropriate facilities for safe receipt, handling, and transhipment of foreign research reactor spent nuclear fuel (Criterion 3);
- The ports must have ready access for intermodal transport (i.e., truck and rail facilities at or close to the selected terminal) (Criterion 4); and
- The human population of the ports and along transportation routes must be low to the extent economically feasible and maximum extent practicable (Criterion 5).

In selecting the final list of seaports from those found acceptable under Criterion 5, DOE applied several desirable port attributes. The potentially most useful of these ports for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel to any of the five DOE management sites, and which also had the highest number of other desirable attributes, were selected for consideration and detailed analysis in the EIS.

D.1.9.1 Criterion 1: Appropriate Port Experience

The first criterion selected is one of the National Defense Authorization Act requirements for using ports with appropriate experience. The criterion is used first because if a port does not currently have appropriate container handling experience, or is unlikely to have this experience during the time period analyzed in this EIS, there is no reason to consider it further. For this screening, commercial ports that handle on the order of at least 20,000 20-ft equivalent units of containerized cargo per year [i.e., any mix of breakbulk, combination breakbulk/container ships, or self-contained ships that are equivalent to unloading (or loading) a small container vessel every week or two] were selected for further detailed analysis under the remaining criteria.

Because containerized spent nuclear fuel requires no special port experience or facilities specific to the handling of radioactive material, ports were not eliminated from consideration because of lack of such experience or facilities.

This criterion excludes experience in handling bulk liquid cargoes (e.g., oil or petrochemicals) or other bulk cargoes (e.g., grain, coal, etc.) unloaded using special cargo equipment not of the type used for receipt and handling of containerized foreign research reactor spent nuclear fuel shipments. It also excludes ports used primarily by fishing fleets or cruise ship liners.

Ports meeting the appropriate experience requirement would be those where port terminal(s) and operators routinely load and/or unload all types of containerized dry cargoes requiring the same type of handling as containerized spent nuclear fuel (e.g., everything from television sets and machine parts to toxic materials, flammable or explosive cargoes, etc.), or are likely to acquire such experience during the time period analyzed in this EIS (i.e., large cargo or container port expansions or improvements are planned within the next several years). DOE found that the status of commercial port facilities is very dynamic and subject to rapid and unpredictable changes. For example, the Port of San Francisco, CA lost four of its five major container lines to the Port of Oakland, CA early in 1994, and the Port of Morehead City, NC, has gone from on the order of 10,000 containers per year a few years ago to essentially no container service at the present time (DOE, 1994d). Similarly, the Port of Richmond, CA (while it still has two container cranes available and acceptable facilities) no longer receives significant numbers of containers (AAPA, 1994), although that could change in the near future.

This criterion also effectively eliminated ports that have infrequent container/breakbulk ship calls, marginal equipment or facilities, and were less likely to have well-trained and experienced personnel than busier ports during the period analyzed in this EIS (adequacy of ports and facilities for receiving, handling, and transhipping such cargoes will be addressed in Section D.1.9.3).

Out of the original list of 153 commercial candidate ports in the contiguous United States that were discussed earlier (excluding the 29 Great Lakes and upper Mississippi River ports), this screening resulted in the identification of 31 candidate seaports (see Table D-2 and Figure D-1). Many of the rejected ports were associated with oil or other bulk shipments, and were not viable for either breakbulk or container operations. These 31 commercial ports are considered to be reasonably representative of the total population of viable commercial seaports in the contiguous United States. Three of the eight military ports evaluated were found to generally satisfy this criterion, allowing for the cyclical nature of military activities at these ports (see Figures D-1 and D-2 and Table D-3). The acceptable military ports included the Military Ocean Terminal Bay Area in Oakland, CA, and the Naval Weapons Station (NWS) in Concord, CA, as potential West Coast ports of entry, and MOTSU for a potential East Coast port of entry. This criterion screened out all naval bases and shipyards in the contiguous United States because they do not regularly handle containerized cargo from ocean-going vessels in any significant quantity.

There is great uncertainty associated with attempts to project the future of port activities and possible availability for receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. Many of the features and facilities of ports addressed in Criterion 3 are inextricably related to the likelihood that any given port will meet the minimum requirements for "appropriate" experience in the future. Thus, for example, if a specific port lacks adequate facilities and equipment at present, and there is no identifiable intention of improving the port in the future, it is unlikely that the port will develop the appropriate

Table D-2 Commercial Ports with Appropriate Experience Receiving, Handling, and Transhipping Containerized Dry Cargoes[a]

U.S. Seaport	Appropriate Experience	U.S. Seaport	Appropriate Experience	U.S. Seaport	Appropriate Experience
Alameda, CA	No	Gloucester City, NJ	No	Port Angeles, WA	No
Albany, NY	No	Gramercy, LA	No	Port Arthur, TX	No
Alexandria, VA	No	Grays Harbor, WA[c]	No	Port Canaveral, FL	No
Anacortes, WA	No	Green Bay, WI	NA	Port Costa, CA	No
Antioch, CA	No	Gulfport, MS	Yes	Port Everglades, FL	Yes
Ashtabula, OH	NA	Hopewell, VA	No	Port Hueneme, CA	No
Astoria, OR	No	Houston, TX	Yes	Port Manatee, FL[c]	No
Baltimore, MD	Yes	Huntington Beach, CA	No	Port Neches, TX	No
Baton Rouge, LA	NA	Huron, OH	NA	Port Royal, SC	No
Bay City, MI	NA	Jacksonville, FL	Yes	Port San Luis, CA	No
Beaumont, TX[c]	No	Kalama, OR	No	Port Sulfur, LA	No
Bellingham, WA	No	Kenosha, WI	NA	Port St. Joe, FL	No
Benicia, CA	No	La Place, LA	No	Port Townsend, WA	No
Boston, MA	Yes	Lake Charles, LA	Yes	Portland, OR	Yes
Bridgeport, CT	No	Long Beach, CA	Yes	Portland, ME[c]	No
Brownsville, TX	No	Longview, WA[c]	No	Portsmouth, NH[c]	No
Brunswick, GA	No	Lorain, OH	NA	Portsmouth, VA	Yes
Buffalo, NY	NA	Los Angeles, CA	Yes	Providence, RI	No
Burns Harbor, IN	NA	Mandalay Beach, CA	No	Raymond, WA	No
Cambridge, MD	No	Manitowoc, WI	NA	Redwood City, CA	No
Camden, NJ[c]	No	Marcus Hook, PA	No	Reedsport, OR	No
Carlsbad, CA	No	Marine City, MI	NA	Reserve, LA	No
Carpinteria, CA	No	Miami, FL	Yes	Richmond, VA	Yes
Charleston, SC	Yes	Milwaukee, WI	NA	Richmond, CA[c]	No
Chicago, IL	NA	Mobile, AL[c]	No	Rochester, NY	NA
Cleveland, OH	NA	Morehead City, NC	No	Sacramento, CA	No
Conneaut, OH	NA	Moss Beach, CA	No	Saginaw, MI	NA
Coos Bay, OR	No	Muskegon, MI	NA	San Diego, CA[c]	No
Corpus Christi, TX[c]	No	New Bedford, MA	No	San Francisco, CA	Yes
Crescent City, CA	No	New Haven, CT	No	Sandusky, OH	NA
Crockett, CA	No	New London, CT	No	Savannah, GA	Yes
Delaware City, DE	No	New Orleans, LA	Yes	Searsport, ME	No
Detroit, MI	NA	New York, NY[b]	Yes	Seattle, WA	Yes
Duluth, MN	NA	Newport News, VA	Yes	Sheboygan, WI	NA
Eddystone, PA	Yes	Newport, OR	No	Stockton, CA	No
Edmonds, WA	No	Norfolk, VA	Yes	Superior, WI	NA
El Segundo, CA	No	Oakland,CA	Yes	Tacoma, WA	Yes
Erie, PA	NA	Ogdensburg, NY	NA	Taft, LA	No
Essexville, MI	NA	Olympia, WA	No	Tampa, FL[c]	No
Estero Point, CA	No	Orange, TX	No	Texas City, TX	No
Eureka, CA	No	Ostrica, LA	No	Toledo, OH	NA
Everett, WA	No	Oswego, NY	NA	Uncle Sam, LA	No
Fairport Harbor, OH	NA	Palm Beach, FL	Yes	Vallejo, CA	No
Fall River, MA	No	Panama City, FL	No	Vancouver, WA[c]	No
Ferndale, WA	No	Pascagoula, MS	No	Venice, LA	No
Fernandina Beach, FL	Yes	Paulsboro, NJ	No	Ventura, CA	No
Freeport, TX	Yes	Pensacola, FL	No	Willapa Bay, WA	No
Friday Harbor, WA	No	Philadelphia, PA	Yes	Wilmington, DE	Yes
Galveston, TX	Yes	Pilottown, LA	No	Wilmington, NC	Yes
Gaviota, CA	No	Pittsburgh, CA	No	Winslow, WA	No
Georgetown, SC	No	Point Wells, WA	No (Closing)		

[a]*For possible use by breakbulk, container, Roll-on/Roll-off or combination vessels. No Great Lakes ports or ports far up the Mississippi River were evaluated in detail because of the unacceptably long transits on crowded inland waterways or need for additional intermodal transfers (listed as "not applicable" or NA).*

[b]*Includes the preferred terminal at Elizabeth, NJ.*

[c]*Does have limited dry cargo facility; could acquire appropriate experience during 10 to 15-years.*

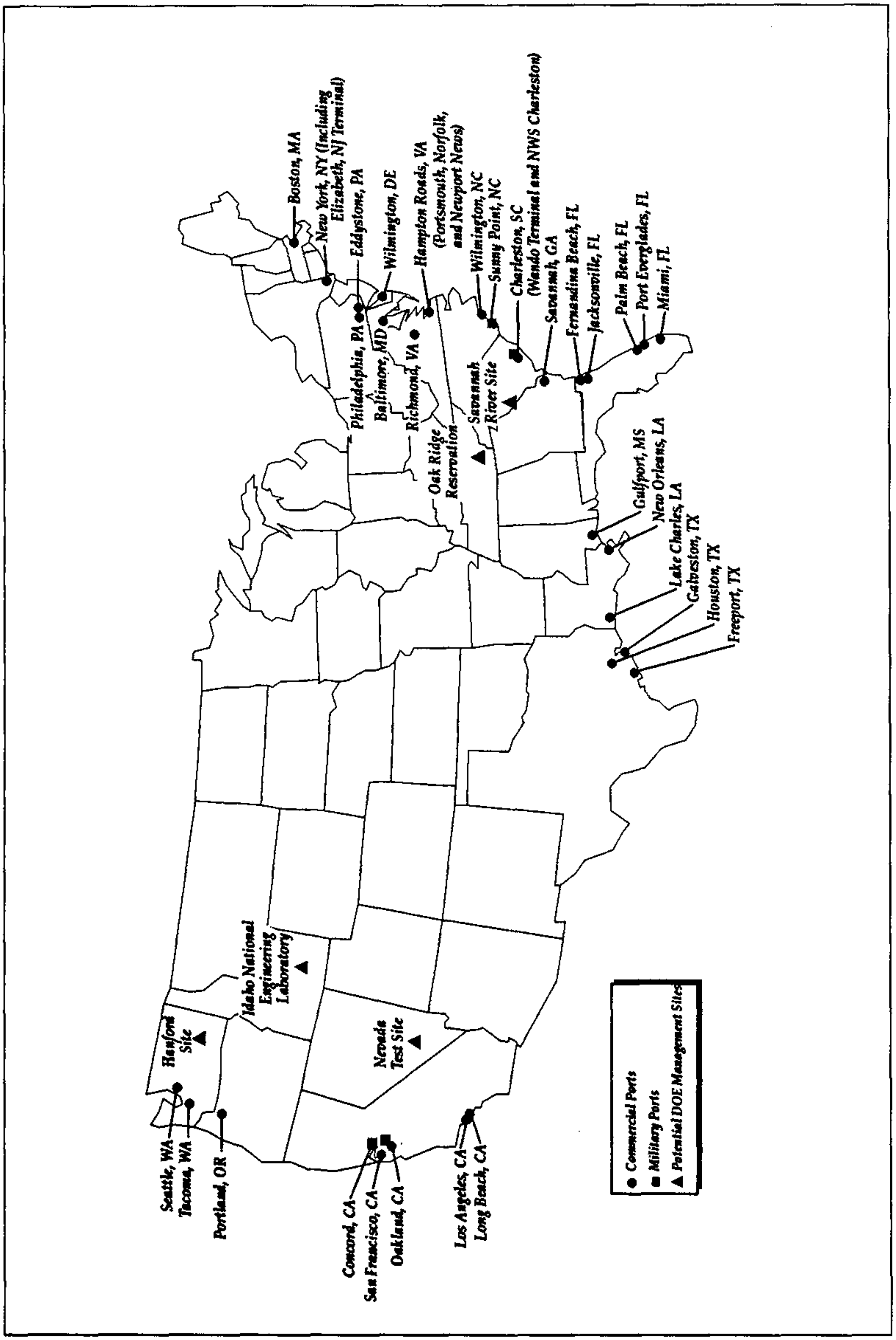


Figure D-1 Locations of Candidate Ports of Entry and DOE Management Sites

❶ *Appropriate Port Experience:*

- Ports routinely handle containerized cargo (at least 20,000 TEUs/yr)

a. *Accepted 31 Commercial Ports:*

Baltimore, MD*
Boston, MA
Charleston, SC
Eddystone, PA*
Elizabeth, NJ
Fernandina Beach, FL
Freeport, TX
Galveston, TX
Gulfport, MS
Houston, TX
Jacksonville, FL
Lake Charles, LA
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA*
New Orleans, LA
Norfolk, VA*
Oakland, CA*
Palm Beach, FL
Philadelphia, PA
Port Everglades, FL
Portland, OR*
Portsmouth, VA*
Richmond, VA
San Francisco, CA
Savannah, GA*
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. *Accepted 3 Military Ports**:*

Military Ocean Terminal
Sunny Point, NC*

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

** Database indicates Port has handled SNF or other Type B cask shipments*

*** Military ports meet 20,000 TEU requirement on a periodic basis, but cycle between high and low work loads based on military demands*

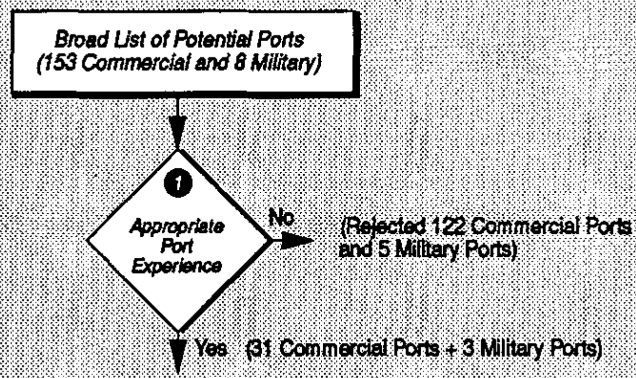


Figure D-2 Screening Ports with Appropriate Experience Criterion

Table D-3 Military Ports with Appropriate Experience Receiving, Handling, and Transhipping Containerized Dry Cargoes

U.S. Military Port	Appropriate Experience[a]
Bayonne, NJ (Military Ocean Terminal)	No
Cheatham Annex, VA (Navy)	No
Concord, CA (Naval Weapons Station)	Yes
Kings Bay, GA (Submarine Base)	No
Oakland, CA (Military Ocean Terminal)	Yes
Port Hueneme, CA (Naval Construction Battalion Center)	No
Sunny Point, NC (Military Ocean Terminal, MOTSU)	Yes
Yorktown, VA (Naval Weapons Station)	No

[a]*Military ports meet 20,000 twenty-foot equivalent units/yr requirement on a periodic basis and retain a corp of experienced port workers.*

Sources of information include: MTMCTEA, 1992 (Bayonne, NJ); MTMC, 1994a (Cheatham Annex, VA); Yocum, 1994a and 1994b (Concord, CA); FHI, 1993b (Kings Bay, GA); MTMC, 1994b; MTMCTEA, 1990 (Port Hueneme, CA); DOE, 1994d (Sunny Point, NC); and FHI, 1994a (Yorktown, VA).

experience required during the period analyzed in this EIS. As a result, DOE searched the available literature and scanned port-specific information from a number of sources (including direct discussions with numerous port officials) to identify planned future port improvements to see if that information could be used to tentatively increase the number of potential ports in subsequent port screenings for other necessary criteria. The results of this review are addressed further in Attachment D1 to this appendix. The basic finding was that all identifiable future port improvements were generally being made in ports that already meet the appropriate experience criterion (also see Attachment D1 to this appendix for projected port improvements that have been identified). As a result, that test for potential future port utilization did not yield any additional ports for subsequent screening.

On the other hand, DOE also found that some ports already have adequate facilities, but for one reason or another have been unable to attract enough container shipments to ensure a reasonable core of experienced workers (e.g., Port of Richmond, CA). In such cases, future projections are extremely uncertain, since there is no way of knowing whether ports will be successful in marketing new business opportunities in the future. As a result, DOE concluded that there was no useful purpose served in keeping these ports in the list for subsequent screening.

D.1.9.2 Criterion 2: Favorable Transit From Open Ocean

This criterion was based on recommendations from the U.S. Merchant Marine Academy Workshop participants who found that a short transit from the open ocean to port was necessary to maximize the safety of shipments of spent nuclear fuel (USMMA, 1994). However, it is clear that since this criterion focuses on ship safety, it is essentially synonymous with the requirement for a favorable ship transit from the open ocean to the port. Thus, while a port might be within a few miles of the open sea, if there were numerous hard shoals, ship wrecks, or reefs along the ship channel, this port might be less desirable than other ports with longer but less risky transits. On the other hand, ports that can only be reached by transporting spent nuclear fuel through long, narrow, winding, or crowded ship channels present additional risks that can be avoided by using ports that are easier to reach.

As a result, DOE concluded that ports meeting the intent of this criterion would have relatively short trips to port from large, deep bodies of water that were either oceans, seas, or notable extensions thereof, such as large bays or sounds (e.g., Chesapeake Bay, San Francisco Bay, or Puget Sound), and which present no special navigational hazards to ships (including adequate width and depth of water in ship channels). A minimum channel depth (mean low water) of 7.6 m (25 ft) was selected to permit use by at least small to intermediate size vessels.

Less desirable were potential ports that could only be reached by traversing long, narrow and/or winding, or crowded ship channels [e.g., the St. Lawrence seaway to a Great Lakes port or the long passage up the Galveston/Houston ship channel to Houston (which is crowded by oil tankers in the channel and numerous petroleum and petrochemical plants along the channel that could impact on ship safety in the event of a plant or pipeline accident)].

Reliable data on risks associated with transits are difficult to find. In 1991, the U.S. Coast Guard established a national database on ship accidents. The 46 Code of Federal Regulations (CFR) §4.05-1 defines reportable accidents as those events that (1) leave a vessel damaged and presenting a navigational hazard (e.g., loss of propulsion or steering) or affect seaworthiness, (2) cause damage in excess of $25,000, or (3) result in serious injury or loss of life. Included in the database are allisions (single ship collisions with fixed structures such as buoys, docks, or bridges), collisions (between two vessels while under power), hard groundings (where a vessel cannot free itself), and fires onboard cargo vessels due to other accidents (USCG, 1994a and 1994b). However, since these accident statistics are not comprehensive and include barge accidents in addition to those involving ocean-going vessels, it is difficult to provide sound and reliable estimates of accident frequencies and types per transit to port.

Using all of the information currently available that pertains to analysis of this criterion, DOE found that the Ports of Richmond VA, New Orleans and Lake Charles, LA, and Houston, TX do not meet the criteria for receipt of foreign research reactor spent nuclear fuel at this time.

As shown in Table D-4 and Figure D-3, application of this criterion resulted in the retention of 27 commercial seaports and three military ports for further analysis.

D.1.9.3 Criterion 3: Appropriate Port Facilities

The National Defense Authorization Act requires the use of ports with "appropriate port facilities" that allow safe handling of foreign research reactor spent nuclear fuel. The U.S. Merchant Marine Academy Workshop recommended as "necessary for safe shipment" that an acceptable port have "adequate port cargo facilities," which included (1) berthing options (e.g., so that conflicting activities at an adjacent berth or onshore could be avoided if necessary), and (2) onsite cranes with trained operators (while it was recognized that ports without cranes could use other means to offload a vessel, the panel preferred ports with cranes).

Thus, port facilities must possess the following minimum physical attributes: (1) adequate water depths alongside piers [at least 7.6 m (25 ft) was selected for this screening] for docking at least small to intermediate-sized vessels, (2) adequate wharfs and quays, with berthing options (in case a potential for conflicting operations exists near the berth of choice), for securing vessels and safely offloading, and carrying the necessary spent nuclear fuel loads, and (3) at least one adequate crane for offloading containerized spent nuclear fuel onto ground transport [at least a 30 metric ton (33 ton) capacity crane was selected for this screening].

While many small ports have cranes with large lift capacities [100 metric tons (110 tons) or more], they are not purpose-built container cranes and must use special container spreaders for use with containers. Although the U.S. Merchant Marine Academy Workshop found it "desirable" for a port to have an adequate purpose-built container crane available, participants determined it was unnecessary to have one.

Table D-4 Required Maritime Transit Criterion for Selection of Seaports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Seaports	Distance from Open Sea (km)[a]	Favorable Transit
Commercial		
Baltimore, MD	240	Yes
Boston, MA	12	Yes
Charleston, SC	11	Yes
Eddystone, PA	120	Yes
Elizabeth, NJ	18	Yes
Fernandina Beach, FL	15	Yes
Freeport, TX	6	Yes
Galveston, TX	16	Yes
Gulfport, MS	30	Yes
Houston, TX	71	No
Jacksonville, FL	11	Yes
Lake Charles, LA	52	No
Long Beach, CA	4	Yes
Los Angeles, CA	5	Yes
Miami, FL	5	Yes
Newport News, VA	40	Yes
New Orleans, LA	160	No
Norfolk, VA	35	Yes
Oakland, CA	15	Yes
Palm Beach, FL	6	Yes
Philadelphia, PA	130	Yes
Port Everglades, FL	2	Yes
Portland, OR	140	Yes
Portsmouth, VA	40	Yes
Richmond, VA	190	No
San Francisco, CA	19	Yes
Savannah, GA	24	Yes
Seattle, WA	5	Yes
Tacoma, WA	5	Yes
Wilmington, DE	100	Yes
Wilmington, NC	38	Yes
Military		
NWS Concord, CA	60	Yes
MOTBA, CA	15	Yes
MOTSU, NC	16	Yes

[a]*To convert distance to miles, divide by 1.6.*

Thus, while it is preferable to avoid any additional risks associated with the use of general purpose cranes (even though small) by using terminals with equipment designed to handle containerized cargo (an alternative to port container cranes might be the use of combination breakbulk/container vessels with shipboard container cranes that are generally operated by trained and experienced port stevedores), a purpose-built container crane was not determined to be necessary to satisfy this criterion. Military ports also represent a special case, since most do not have such purpose-built container cranes, and use a container spreader attachment when necessary.

❷ *Favorable Transit to Port:*

- Port within reasonable distance from the open sea, with favorable transit

a. *Accepted 27 Commercial Ports:*

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Freeport, TX
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Palm Beach, FL
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. *Accepted 3 Military Ports:*

Military Ocean Terminal
Sunny Point, NC

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

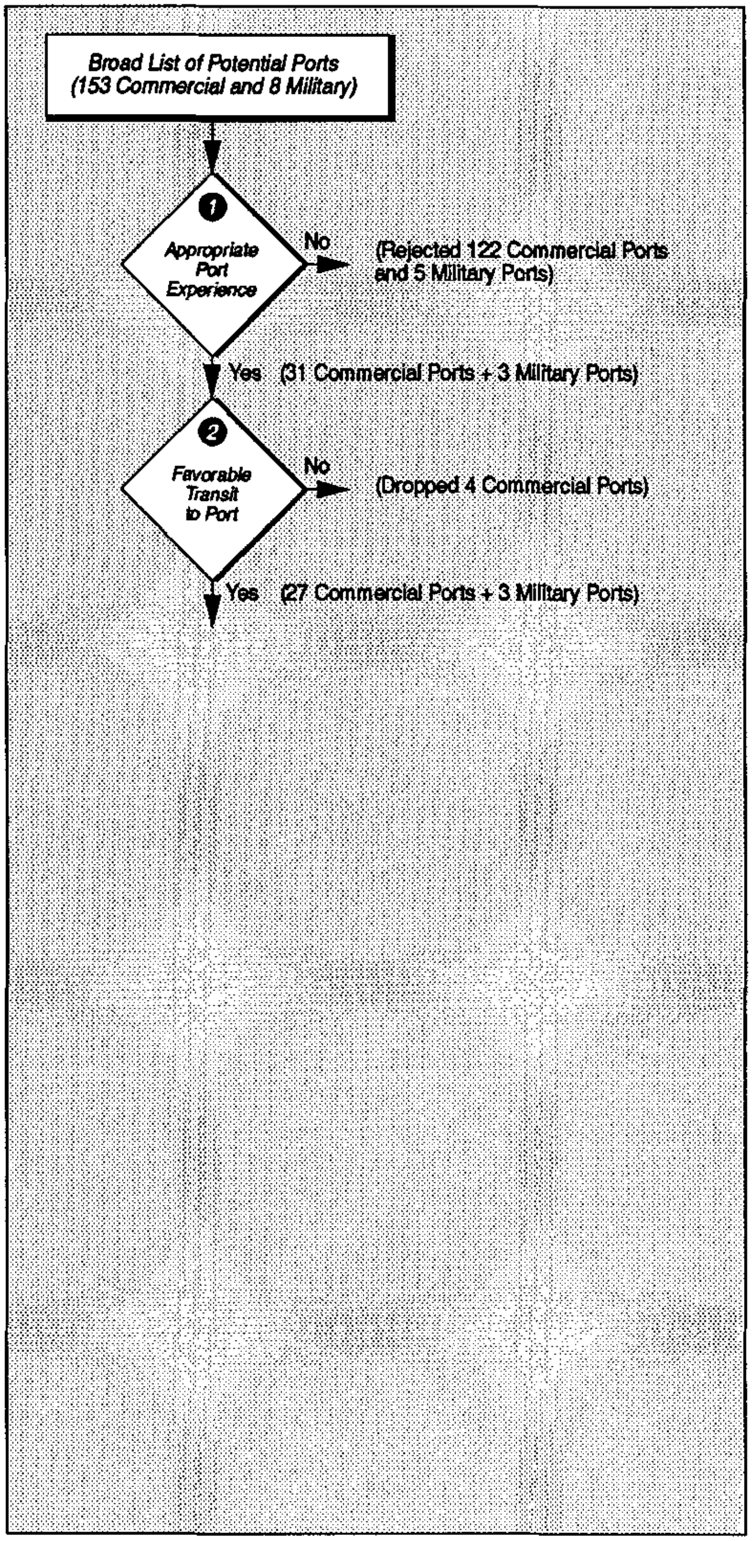


Figure D-3 Screening Ports for Favorable Transit Criterion from Sea to Port

Because containerized spent nuclear fuel requires no special port experience or facilities specific to the handling of radioactive material, ports were not eliminated from consideration because of lack of such experience or facilities.

As noted earlier, there is no reasonable way of determining the future likelihood that currently marginal ports that already have adequate facilities (but simply lack "appropriate experience") will acquire such experience. It depends totally on whether the ports will be able to induce shipping lines to use their facilities. In the area of appropriate facilities, however, there is much less uncertainty in making such determinations, since the planning process for port improvements must be made years in advance in order to allow time for land acquisition, funding, and other approvals before such improvements can be made. Therefore, those ports that have current plans for improvements that might permit their consideration for purposes of this EIS are much easier to identify. As a result, available information relating to future port improvements was studied carefully. Ports with substantial identified improvements or developments during the period analyzed in this EIS include: Baltimore, MD; Boston, MA; Charleston, SC; Fernandina Beach, FL; Gulfport, MS; Jacksonville, FL; San Francisco, CA; Oakland, CA; Long Beach, CA; Naval Weapons Station Concord, CA; Los Angeles, CA; Miami, FL; Mobile, AL; New Orleans, LA; Norfolk, VA; Philadelphia, PA; Port Everglades, FL; New York, NY; Portland, OR; Savannah, GA; Seattle and Tacoma, WA; and Wilmington, DE. Details on these improvements are shown in Attachment D1 to this appendix. All of these ports (except Mobile, AL) currently have both adequate experience and facilities without further improvements. Therefore, no additional ports were identified for foreign research reactor spent nuclear fuel receipt in the future. (Mobile, AL will meet the requirement for experience if it approximately doubles its current container business in the future, but that is too speculative to be useful at this time).

In addition to physical attributes of port facilities, public safety also depends on the reliability of the personnel operating the facilities. In addition to port accidents related to a failure of the container-handling equipment, human error can also increase risks of accidents. The U.S. Merchant Marine Academy Workshop identified the skill of the labor force at a port as an important (but not mandatory) criterion.

While the U.S. Merchant Marine Academy Workshop preferred ports with cranes, it also considered the use of roll-on/roll-off vessels, but preferred the use of other vessels for a number of reasons. Although roll-on/roll-off is not as likely to be used for spent nuclear fuel shipments as conventional cargo vessels due to costs, availability, and other factors, such vessels require only adequate water depths and appropriate piers to receive foreign research reactor spent nuclear fuel. As a result, the presence of roll-on/roll-off facilities is noted in Section D.2 in the detailed discussions of potential ports, but was not considered an adequate sole basis for port selection.

Several related "desirable" attributes for port facilities were recommended by the U.S. Merchant Marine Academy Workshop or identified during the port analyses. These attributes which would contribute to a port having "appropriate facilities," but were not required for safe receipt, handling, and transhipment of spent nuclear fuel include: (1) secure short-term storage areas (in the event of unexpected events such as snow or icing of roads), (2) the existence of emergency planning and training, (3) the absence of environmentally sensitive areas in port or local restrictions on movement of spent nuclear fuel, (4) the absence of conflicting uses (e.g., explosives, petroleum, tourism), and (5) minimal likelihood of severe natural phenomena impacting port activities (such as high winds from hurricanes, floods, earthquakes, volcanoes). These desirable port attributes are used in final port selection in Section D.1.8.5, and are discussed for each port in Section D.2.

In applying the DOE criteria, it became evident that the majority of the ports that met the first required screening criterion (Appropriate Port Experience) also met these requirements. Application of the Appropriate Facility Criterion retained 25 commercial ports and three military seaports for further analysis. Two commercial ports, Freeport, TX and Palm Beach, FL were dropped due to the application of this criterion. The results of this screening are summarized in Figure D-4.

D.1.9.4 Criterion 4: Ready Access to Intermodal Transportation

A U.S. Merchant Marine Academy Workshop criterion determined to be necessary for safe shipment of spent nuclear fuel was "intermodal access", which means "ready access from a port" to truck and rail routes. It is becoming common practice for ports with intermodal transfer facilities to carry off-loaded containers on special port-owned container handling equipment to a marshalling yard adjacent the terminal, where the containers are loaded onto trucks or rail for shipment to the consignee. Such transfers tend to minimize traffic congestion at shipside by using experienced port personnel and specialized port equipment. These intermodal transfers are increasingly accomplished with purpose-built container handling equipment (straddle carriers, sidelifts, front-end loaders, stackers, and container forklifts) that require no additional workmen for container handling. Given that the handling is done very rapidly and securely by a single operator, the opportunities for additional worker exposures or serious accidents are minimized. Moving a container from a pier to a marshalling area a few kilometers more distant than one at a terminal has little significance with regard to either worker exposure or public risk (see, for example, container handling equipment in Jane's, 1992).

DOE found that most of the potential ports accepted under the three preceding required criteria also had good access to interstate highways and rail transport. However, smaller ports and most military ports with more limited facilities could also accept containerized spent nuclear fuel from combination breakbulk/container vessels or roll-on/roll-off vessels. These ports often have limited intermodal capabilities for rail in the immediate vicinity of a pier, but the spent nuclear fuel could be trucked to a rail area (often a few miles or less) for loading on a railcar. While these arrangements could involve an additional intermodal transfer, such transfers are typically also done rapidly using special container handling equipment. Therefore, they do not involve significant additional opportunities for worker exposure or accidents than would be the case for movement of foreign research reactor spent nuclear fuel from a pier to an intermodal yard at a large port.

DOE concluded that the lack of an intermodal rail facility immediately at a terminal should not eliminate an otherwise desirable port from further evaluation, if rail access was reasonably close to the port (e.g., container cargo from the Wando Terminal in Charleston, SC, must be trucked a few miles to an intermodal facility in North Charleston for transfer to rail). All ports evaluated have acceptable intermodal access for trucks, although smaller ports typically do not have dedicated truck routes for access to interstates, and may require short transports through sometimes congested local traffic to reach the interstate highway system. This apparent conflict between requirements for ready intermodal access at ports and the National Defense Authorization Act requirement for using ports with the "lowest human populations" has been balanced to permit some small ports with more limited intermodal capabilities to be considered for further screening, since the additional public impacts associated with a few miles' transport through urban populations would be small compared to public impacts associated with transport over hundreds or thousands of miles of the country's Interstate highway system.

Application of the intermodal access criterion resulted in acceptance of 25 commercial seaports and three military ports (i.e., no additional ports were rejected) for further analysis using the remaining DOE criteria. The results of applying this criterion to commercial and military ports are shown in Figure D-5. Details regarding intermodal access are addressed in each port description in Section D.2.

❸ *Appropriate Facilities:*

- Adequate crane(s), piers, depth of water alongside, etc.

a. Accepted 25 Commercial Ports:

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. Accepted 3 Military Ports:

Military Ocean Terminal
Sunny Point, NC

Military Ocean Terminal
Oakland, CA

Naval Weapons Station
Concord, CA

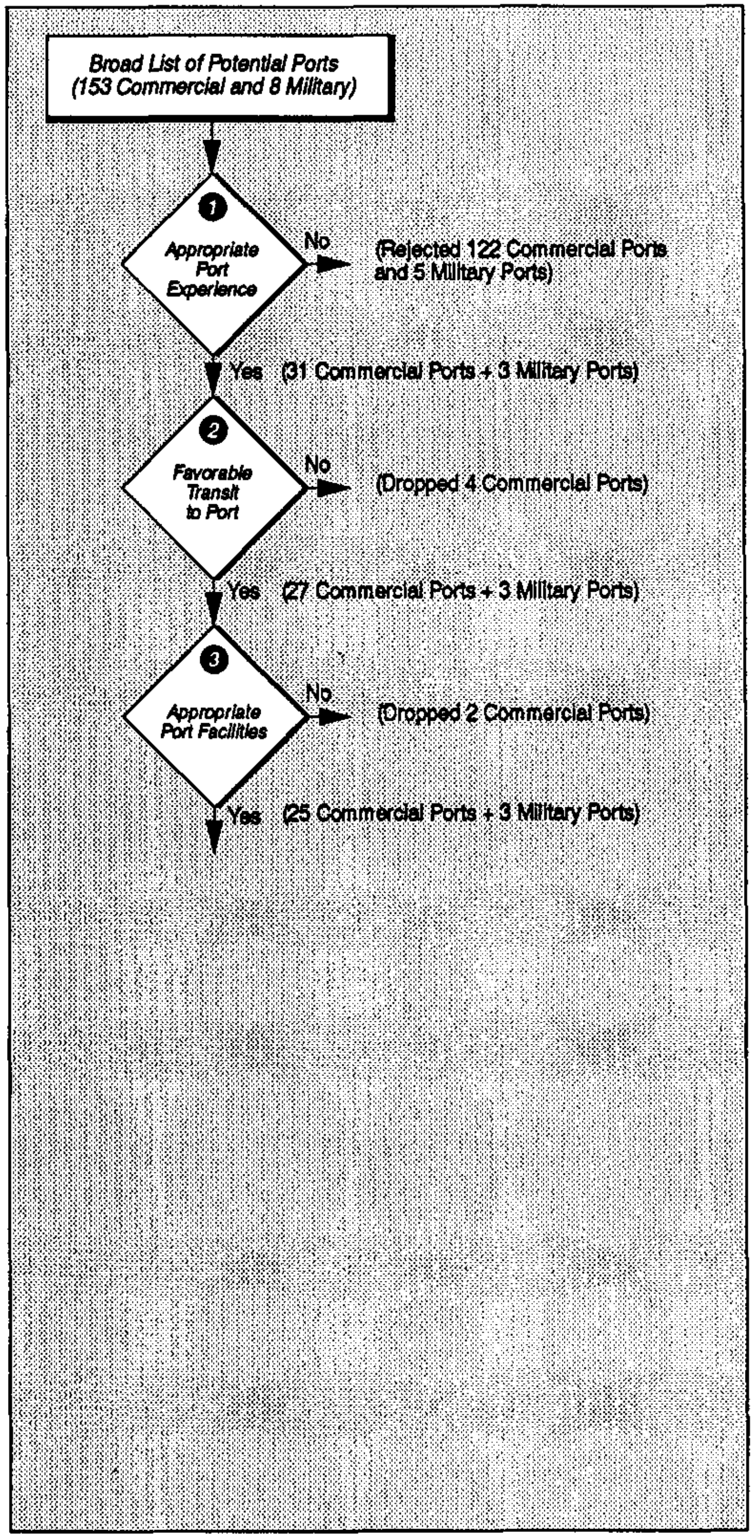


Figure D-4 Screening Ports with Appropriate Facilities Criterion

❹ ***Intermodal Access:***

a. ***Accepted 25 Commercial Ports (SRS and ORNL unless otherwise specified):***

Baltimore, MD
Boston, MA
Charleston, SC
Eddystone, PA
Elizabeth, NJ
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Long Beach, CA
Los Angeles, CA
Miami, FL
Newport News, VA
Norfolk, VA
Oakland, CA
Philadelphia, PA
Port Everglades, FL
Portland, OR
Portsmouth, VA
San Francisco, CA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. ***Accepted 3 Military Ports:***

Military Ocean Terminal Sunny Point, NC

Military Ocean Terminal Oakland, CA

Naval Weapons Station Concord, CA

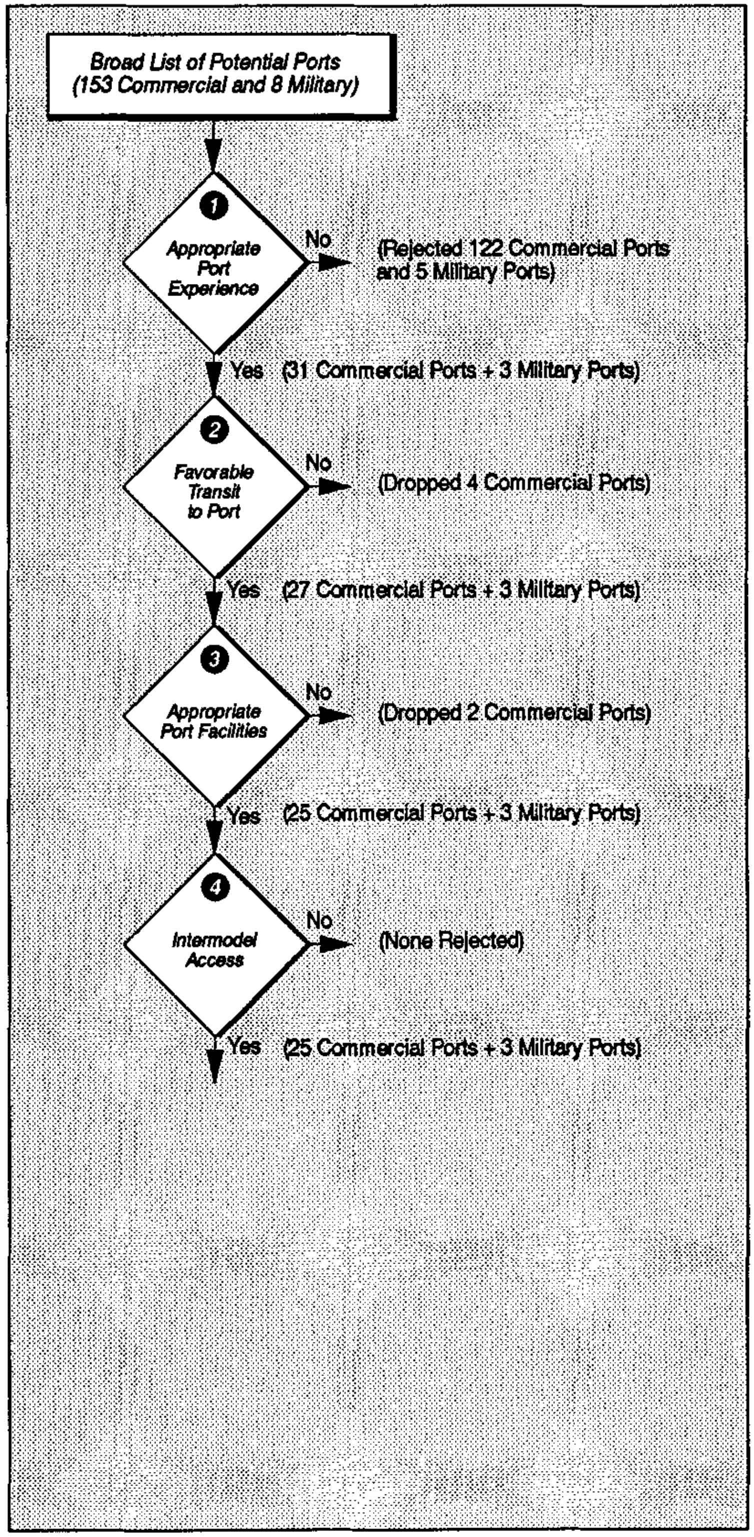


Figure D-5 Screening Ports for Ready Access to Intermodal Transportation

D.1.9.5 Criterion 5: Human Populations

While only dealing with foreign research reactor spent nuclear fuel shipments bound for the Savannah River Site, the Federal court ruling discussed in Section D.1.3 indicates that the courts consider port populations to be an important ingredient in the National Environmental Policy Act (NEPA) process for assessing the range of reasonable port alternatives.

NEPA requires that DOE consider a range of reasonable alternatives for potential ports of entry. On the other hand, the National Defense Authorization Act port selection factors required that, if economically feasible and to the maximum extent practicable, ports of entry for foreign research reactor spent nuclear fuel bound for the Savannah River Site have the lowest human populations in the area surrounding the port. While the National Defense Authorization Act was written specifically to regulate the receipt and storage of foreign research reactor spent nuclear fuel at the DOE's Savannah River Site, DOE elected to apply this criterion in identifying ports of entry for all five potential sites, to the maximum extent practicable.

DOE has considered a number of potential definitions of "lowest human populations" and resulting models that might be used to satisfy the National Defense Authorization Act lowest population factor (NDAA, 1993). These include using the same approach used in the Urgent Relief Environmental Assessment (DOE, 1994d), and variations that might be useful for identifying ports for inclusion in this EIS. A description of the various approaches that were considered are provided in Attachment D3 to this appendix.

As shown in the Urgent Relief Environmental Assessment (DOE, 1994d) and this EIS (Chapter 4), public risk is driven not only by port populations, but by the populations within the immediate proximity of truck and rail shipments from each port to each management site. For each selected port and each selected mode of overland transport (truck or rail), the total "affected" population represents a unique population surrounding the port plus those along the transport route to each of the five potential DOE management sites. DOE considered the affected populations outside the immediate port vicinity along the routes to the management sites to be as important for protection of public health and safety as those within the vicinity of the port terminals, for both incident-free transport and a range of potential accidents.

DOE evaluated port populations within the radii of three distances: 1.6 kilometers (km) [1 mile (mi)], 8.0 km (5 mi), and 16 km (10 mi). These populations are shown in Table D-5. DOE expects that the 1.6 km (1 mi) radius population would include resident members of the public immediately outside the port who would be the most likely to be affected by severe accidents and incident-free impacts. In addition, the radioactivity, which is hypothesized to be released from a very severe accident (long-term fire leading to severe cask damage), would be lofted high into the air and would not normally produce peak ground-level air concentrations until well outside the 1.6 km (1 mi) radius. Therefore, the 1.6 km (1 mi) population was not considered adequate to reflect the population criterion.

The population within a 16 km (10 mi) radius was selected to be consistent with the results of analyses of severe hypothetical accidents described in Section D.5 of this appendix. For severe accidents in ports, the maximum radiation dose to an individual located 16 km (10 mi) from the port is typically much lower than the dose to the maximally exposed individual. However, analyses of the potential impacts of severe accidents in a range of port populations show that the average dose to members of the public within a 16 km (10 mi) radius of the port is higher than the average per capita dose for any of the larger radii around the port for typical (i.e., 50th percentile) meteorology and typical dry deposition and fallout patterns. Further, as discussed in section D.5.4, most of the population dose for even severe accidents occurs within the 16 km (10 mi) radius. Less energetic accident scenarios would cause less dispersion and

Table D-5 Total Populations within Three Distances of Selected U.S. Ports

Name of the Port	*Within 1.6 km (1 mi)*	*Within 8 km (5 mi)*	*Within 16 km (10 mi)*
NWS Concord, CA	14	71,152	381,070
MOTSU, NC	21	960	7,995
Tacoma, WA	94	172,124	511,575
Portland, OR	280	69,039	356,064
Elizabeth, NJ	378	596,076	3,223,038
MOTBA, CA	419	312,133	1,288,699
Jacksonville, FL	523	72,313	334,212
Seattle, WA	557	270,145	753,296
Wilmington, DE	753	166,165	381,502
Gulfport, MS	761	50,218	113,153
Baltimore, MD	818	352,730	1,182,024
Savannah, GA	860	30,845	155,166
Long Beach, CA	1,025	270,336	1,014,418
Charleston, SC	1,550	81,874	233,424
Oakland, CA	1,901	296,661	1,387,611
Miami, FL	2,043	251,551	833,057
Fernandina Beach, FL	2,086	11,787	32,952
Portsmouth, VA	2,554	269,314	665,700
Newport News, VA	2,637	86,993	430,757
Wilmington, NC	2,690	60,308	115,057
Los Angeles, CA	2,918	362,397	1,124,493
Norfolk, VA	2,982	227,290	681,864
Boston, MA	3,084	495,679	1,466,233
Port Everglades, FL	3,927	175,320	714,176
Philadelphia, PA	5,878	50,687	1,915,775
Eddystone, PA	6,179	204,969	827,564
Galveston, TX	8,115	49,175	73,322
San Francisco, CA	9,671	592,869	1,265,529

even smaller doses beyond 16 km (10 mi). Therefore, DOE selected the 16 km (10 mi) radius population to represent the port populations most likely to be impacted by both incident-free transport and the entire range of potential port accidents.

It should be noted that while the populations within the 16 km (10 mi) radius include the populations within 0.8 km (0.5 mi) of the transportation route out to 16 km (10 mi) from the port and result in some double-counting of populations, the results provide only somewhat conservative estimates of the total affected population for each port/management site combination considered.

The populations along truck and rail routes are those computed in Appendix E for the transportation analysis impacts for incident-free transportation.

In summary, this evaluation considered the following population factors:

- Total 1990 Census population within a 16 km (10 mi) radius of the port facilities, and
- Total 1990 Census population within 0.8 km (0.5 mi) of the transport route that would be exposed during transport (from each port to each of the potential DOE management sites).

The statistical distribution of these combined populations for truck transport is shown in Figure D-6. The distribution exhibits some skewing due to a few very large port/site populations, such as around Boston, MA and Elizabeth, NJ. The statistical distribution of combined populations for rail transport is shown in Figure D-7, and again exhibits some skewing due to a few very high population ports. These port/site populations are not clearly normal and are better fit by a Poisson (so-called rare event) distribution, which is often the case for small sample sizes. However, for purposes of developing a systematic and fair method (i.e., one with minimal subjectivity) for evaluating port/site populations, DOE assumed, given the large uncertainty and variances for the small sample sizes for each port/site combined population, that the combined populations for truck transit and the combined populations for rail transit are approximately normal. The port/site population distributions for each of the five management sites (truck and rail routes) are shown in Figures D-8 through D-17, with the bounds associated with the mean plus and minus one standard deviation marked for reference.

For purposes of identifying an acceptable range of ports of entry for the receipt of foreign research reactor spent nuclear fuel, DOE assumed that port/site population combinations greater than approximately one standard deviation above the mean would not be desirable (i.e., about 84 percent of the port/site populations would exhibit statistically lower populations). Thus, the range of ports would include most of the 28 ports being considered, but avoid the extremely large populations around Boston, MA, Elizabeth, NJ, and Philadelphia, PA.

From the remaining 25 ports, DOE assumed that population combinations below the mean combined population would meet the low population criterion while combined populations above the mean would not. As seen in Figures D-8 through D-17, some unique port/site populations would be acceptable for several potential management sites, while other populations would have very limited utility. The potential usefulness of low population ports in relation to this EIS is addressed in Section D.1.9.6. This screening would result in the elimination of an additional five commercial ports and one military port from the list. These commercial ports are Baltimore, MD, and Long Beach, Los Angeles, Oakland, and San Francisco, CA. The military port is Military Ocean Terminal Bay Area in Oakland, CA. The results of the population screening are summarized in Figure D-18.

As previously discussed, the position of maritime experts (USMMA, 1994) is that all of the ports evaluated under the DOE-developed criteria for populations could safely receive and tranship foreign research reactor spent nuclear fuel to all five of the potential DOE management sites. Further, the EIS analyses show that the conservatively calculated impacts would be extremely low. The identification of a smaller number of preferred ports of entry is driven by the requirements of the National Defense Authorization Act, not by any significant safety issues.

As promised in the Urgent Relief Environment Assessment, DOE has also considered future population growth near potential port facilities over the time period considered in this EIS. Year 2010 estimates of projected growth from the 1990 census populations were provided by the states hosting the selected ports and other sources where necessary. Population growth patterns in port cities are continuously changing in ways that cannot be accurately forecast 10 or more years into the future. Nevertheless, the projected port populations based on these growth factors were scrutinized to be sure that no unacceptably large growth would occur around the list of ports selected under the DOE "lowest human population" criterion. The port growth factors used for projecting potential future impacts of port accidents are summarized in Attachment D2 to this appendix, and were used to make final port selections, where appropriate, as discussed in the next section.

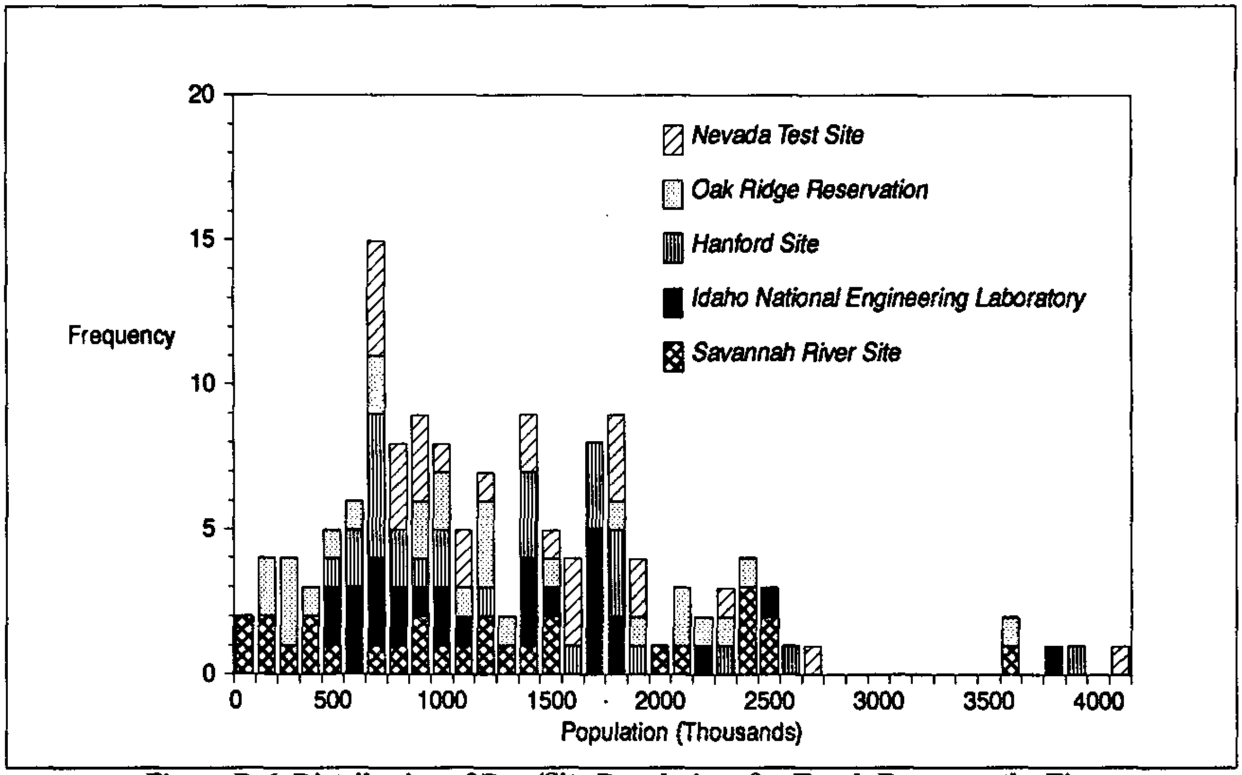


Figure D-6 Distribution of Port/Site Populations for Truck Routes to the Five Management Sites

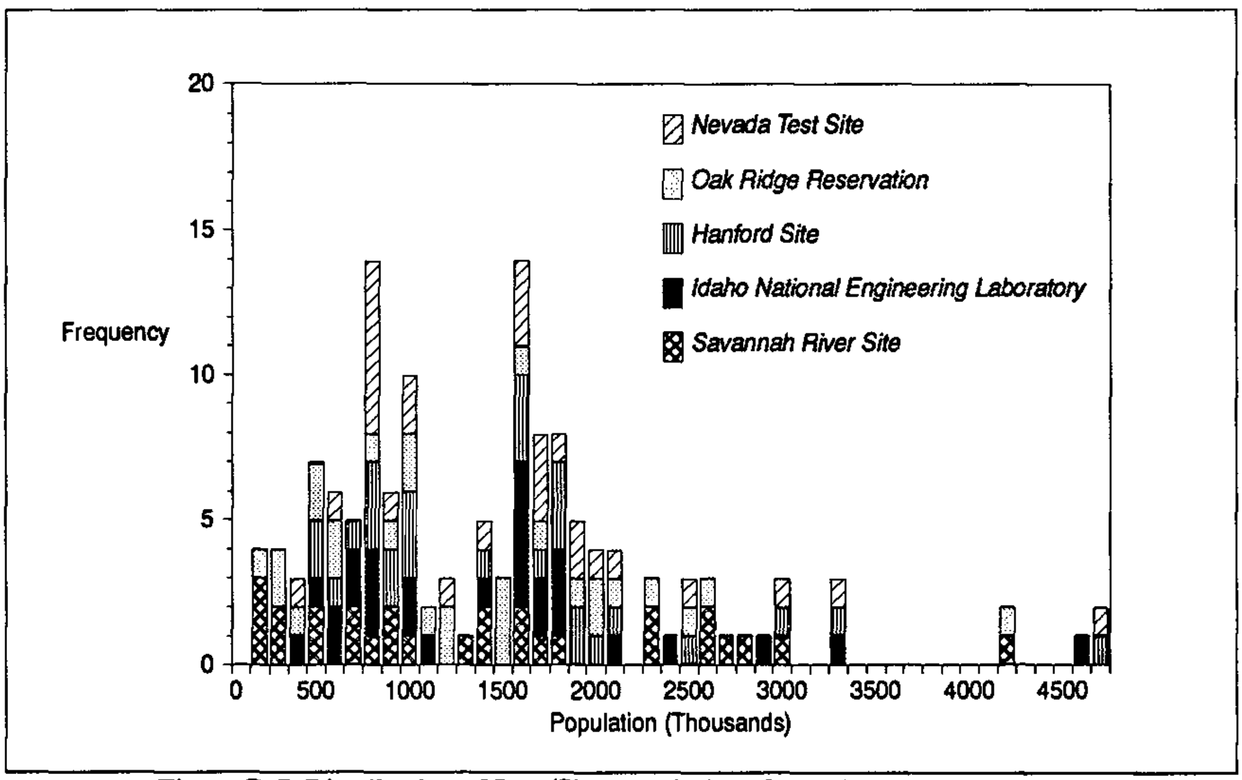


Figure D-7 Distribution of Port/Site Populations for Rail Routes to the Five Management Sites

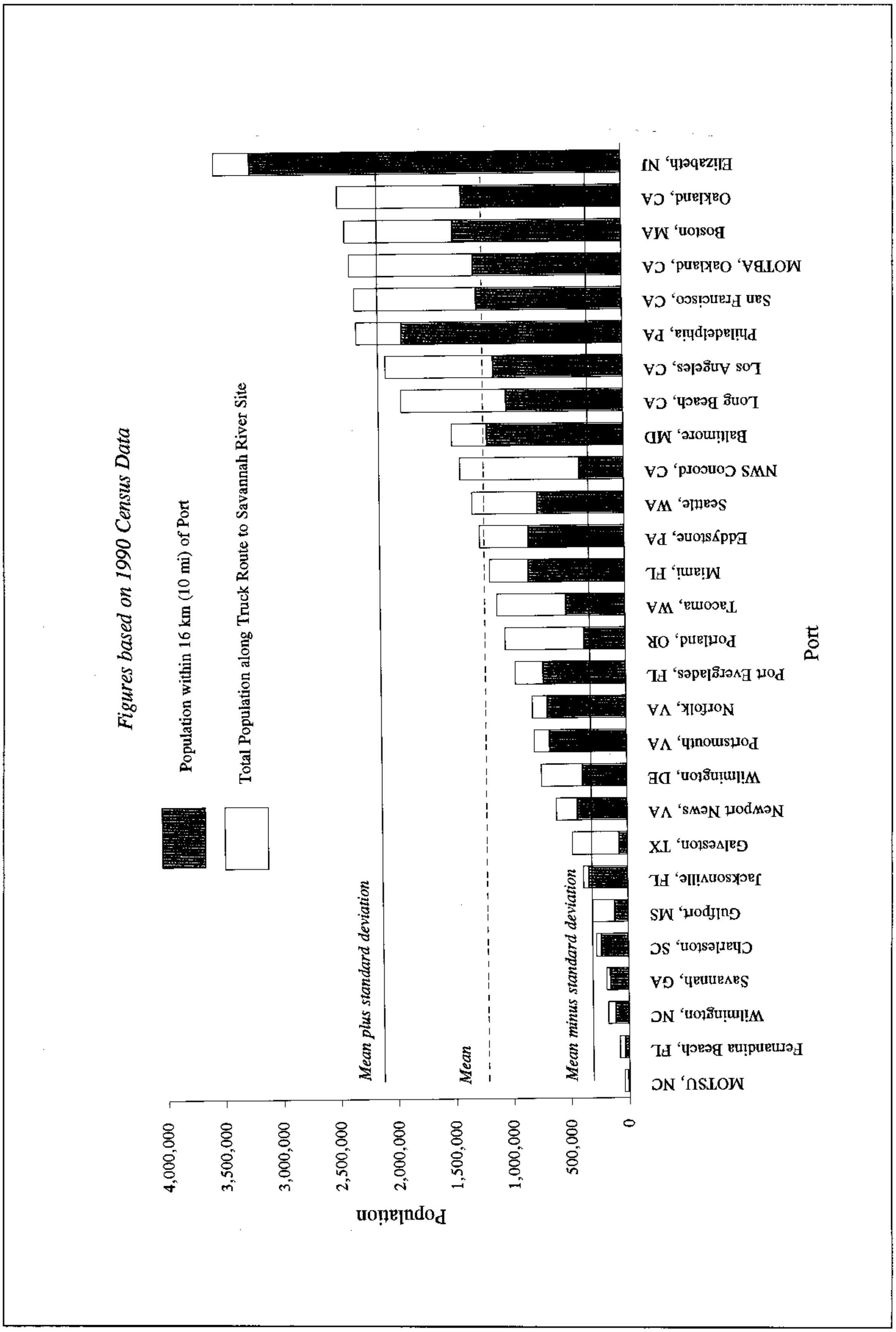


Figure D-8 Population Distribution for Savannah River Site by Truck

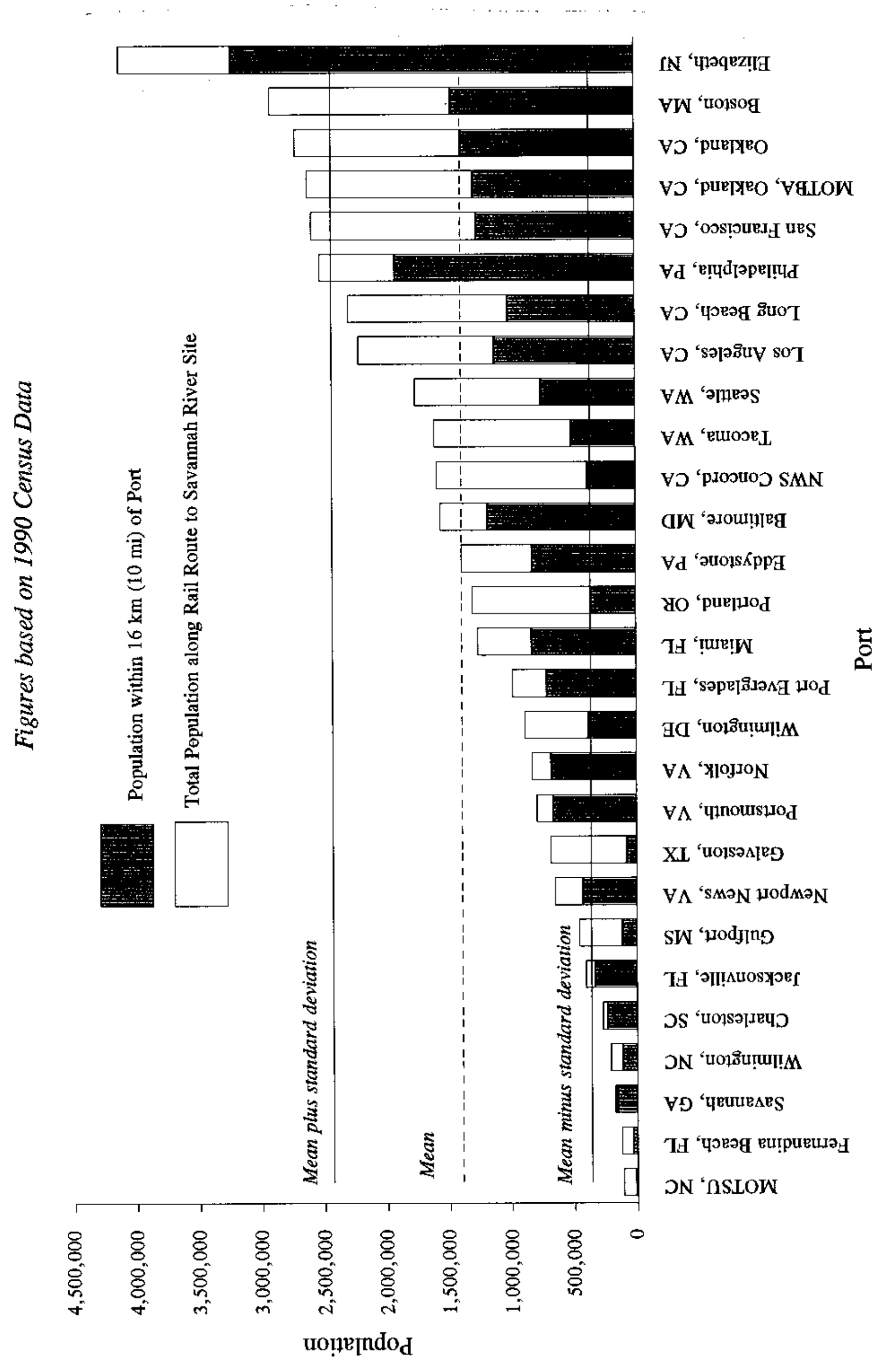


Figure D-9 Population Distribution for Savannah River Site by Rail

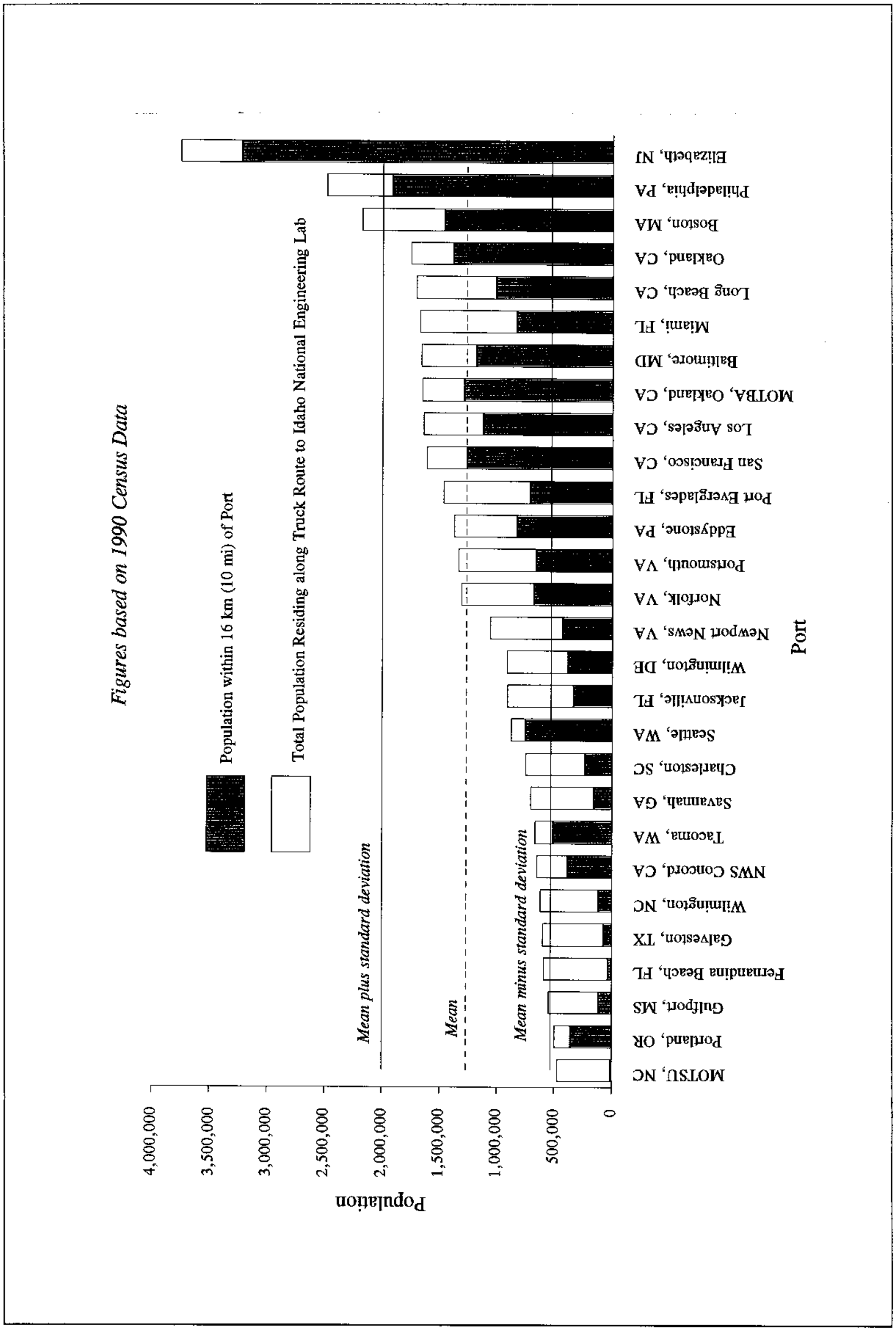


Figure D-10 Population Distribution for Idaho National Engineering Laboratory by Truck

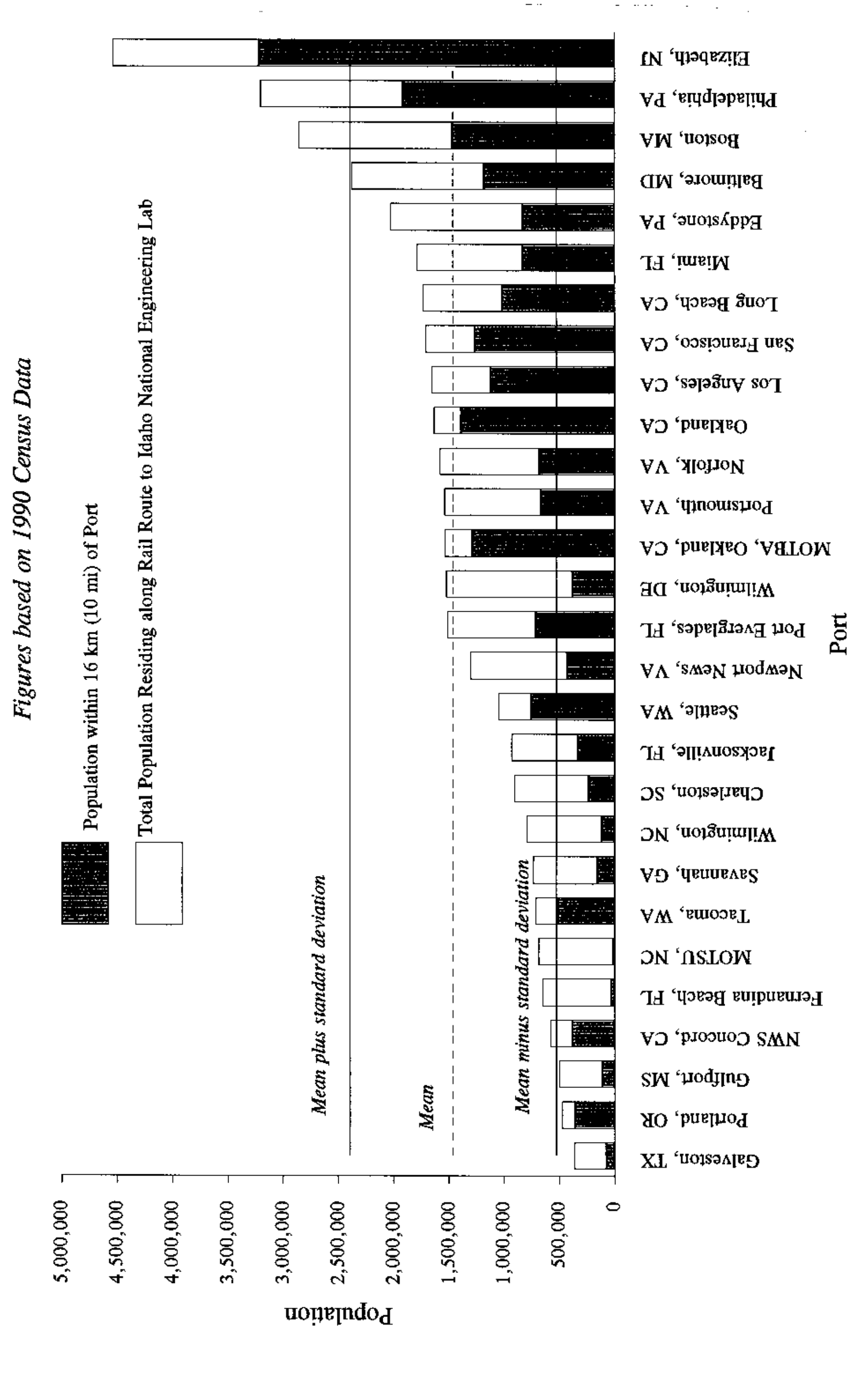


Figure D-11 Population Distribution for Idaho National Engineering Laboratory by Rail

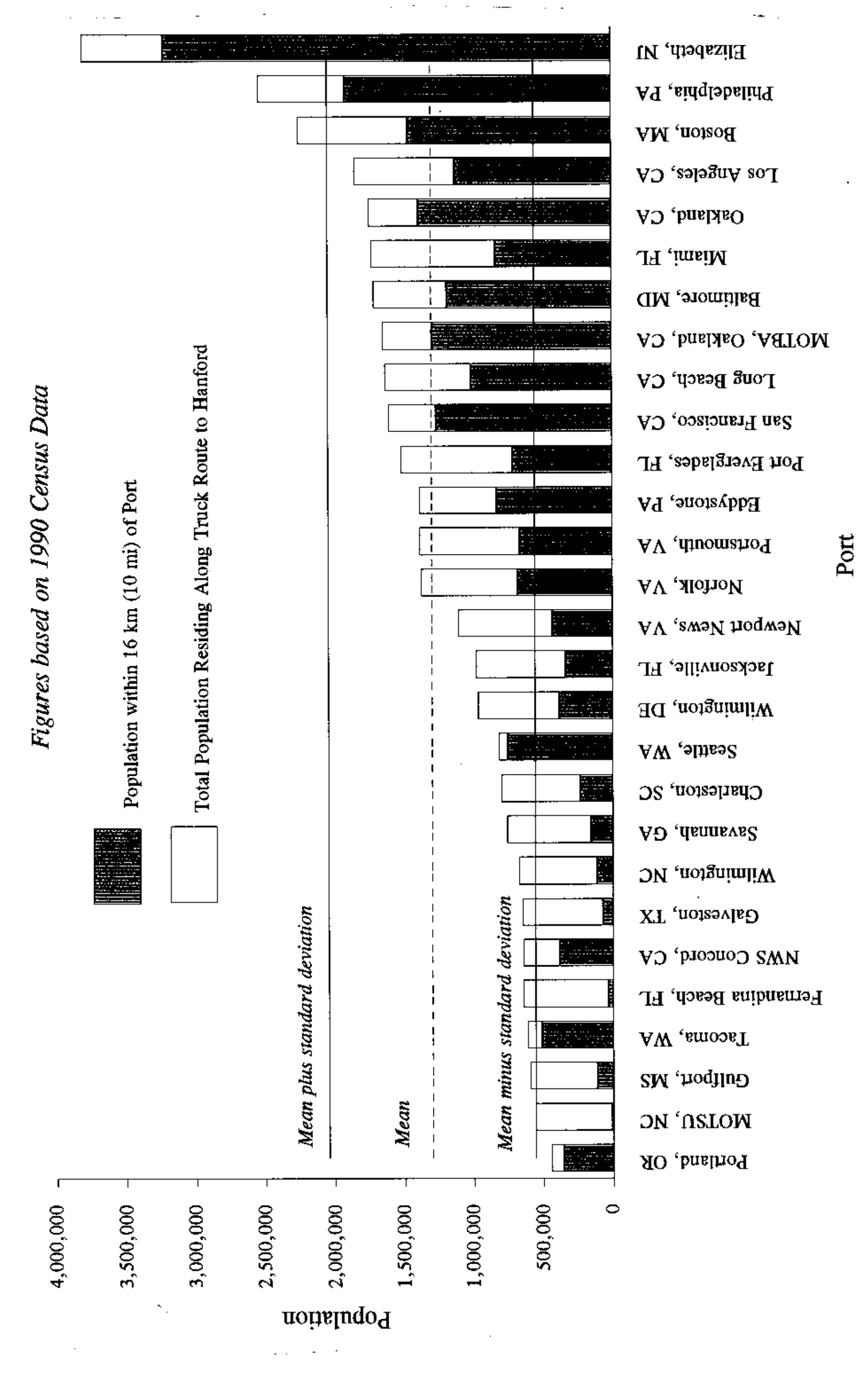


Figure D-12 Population Distribution for Hanford Site by Truck

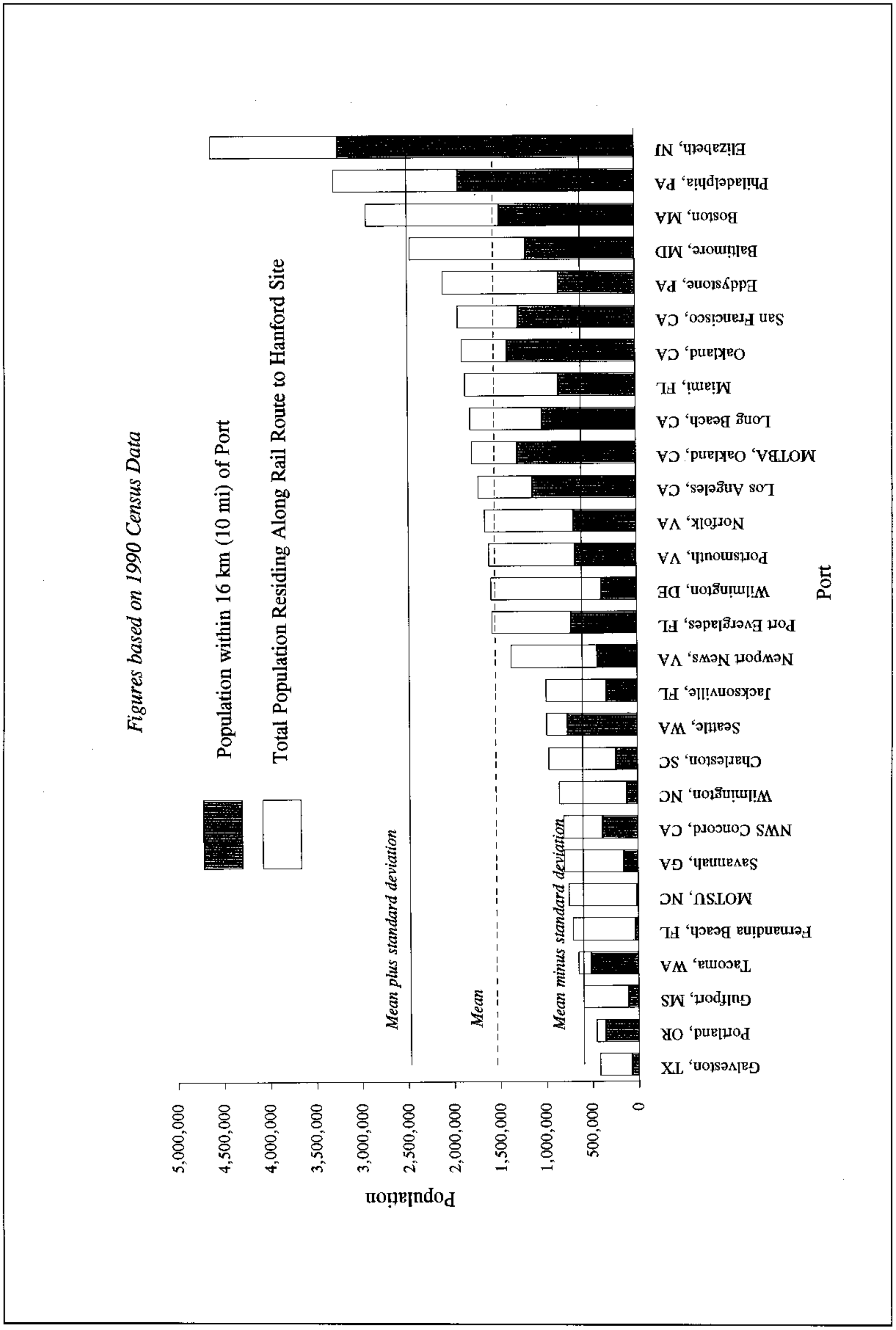


Figure D-13 Population Distribution for Hanford Site by Rail

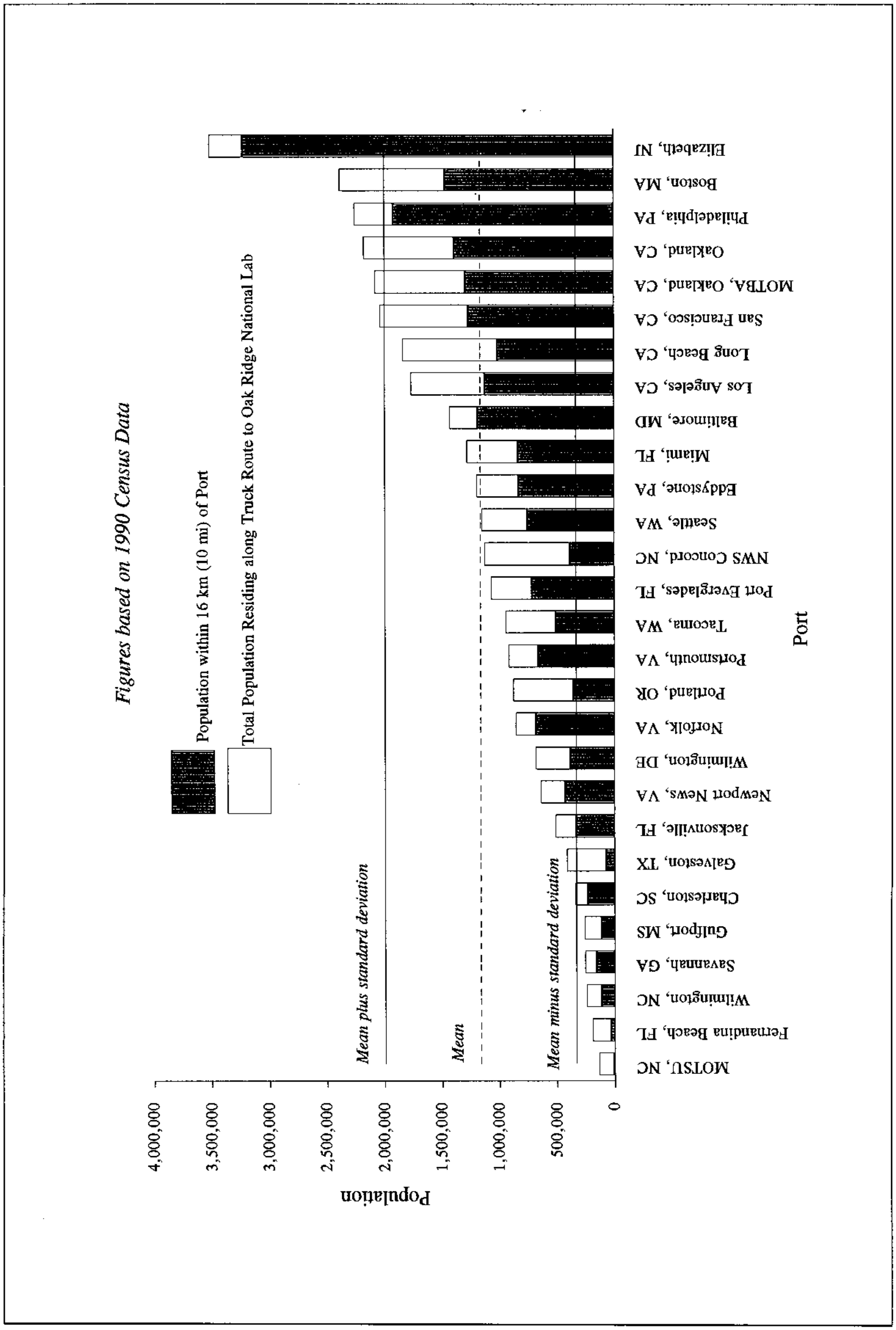


Figure D-14 Population Distribution for Oak Ridge Reservation by Truck

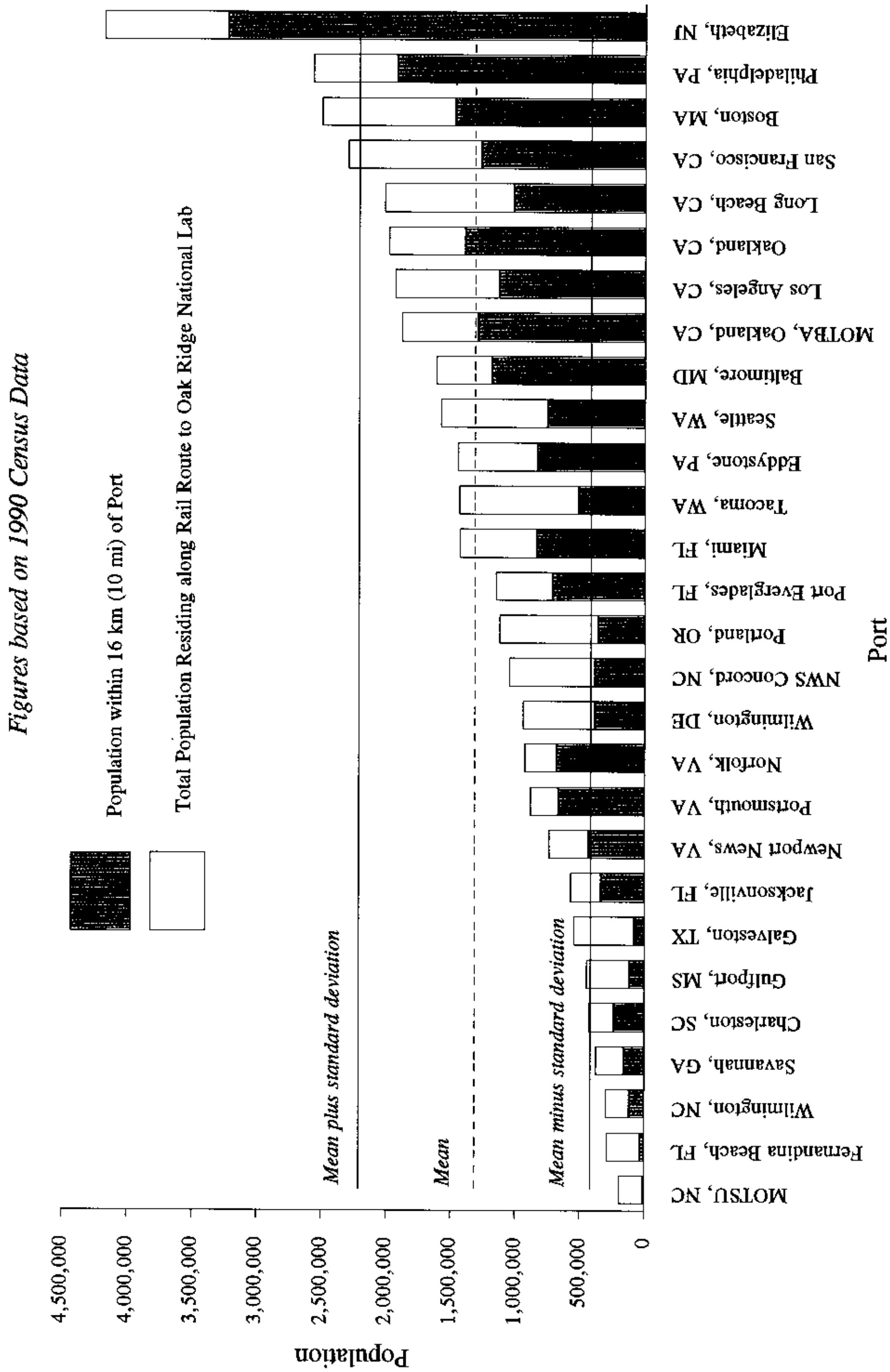


Figure D-15 Population Distribution for Oak Ridge Reservation by Rail

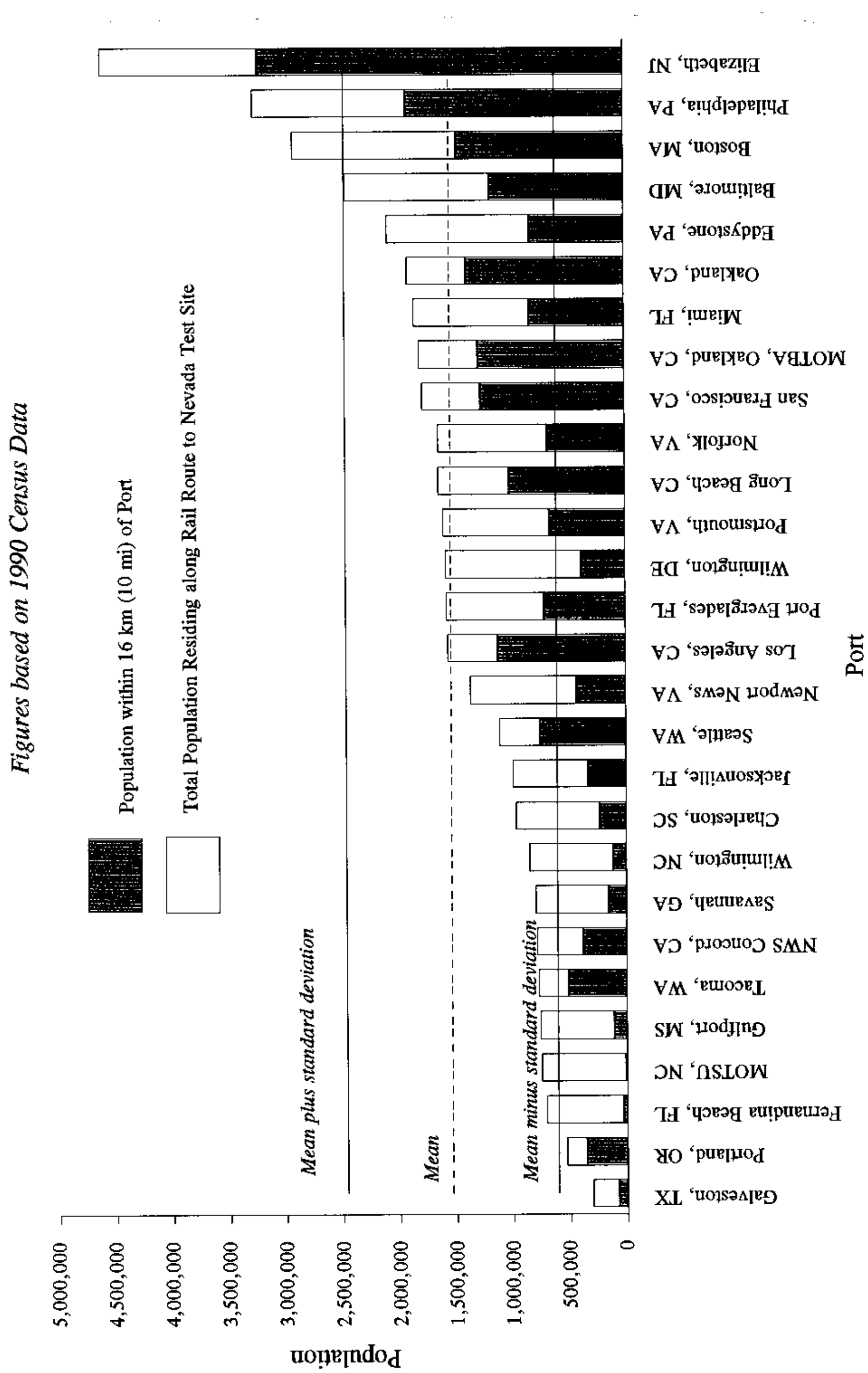


Figure D-17 Population Distribution for Nevada Test Site by Rail

❺ *Low Population Criteria:*

a. ***Accepted 17 Commercial Ports:***

Charleston, SC
Eddystone, PA
Fernandina Beach, FL
Galveston, TX
Gulfport, MS
Jacksonville, FL
Miami, FL
Newport News, VA
Norfolk, VA
Port Everglades, FL
Portland, OR
Portsmouth, VA
Savannah, GA
Seattle, WA
Tacoma, WA
Wilmington, DE
Wilmington, NC

b. ***Accepted 2 Military Ports:***

Military Ocean Terminal
Sunny Point, NC
Naval Weapons Station
Concord, CA

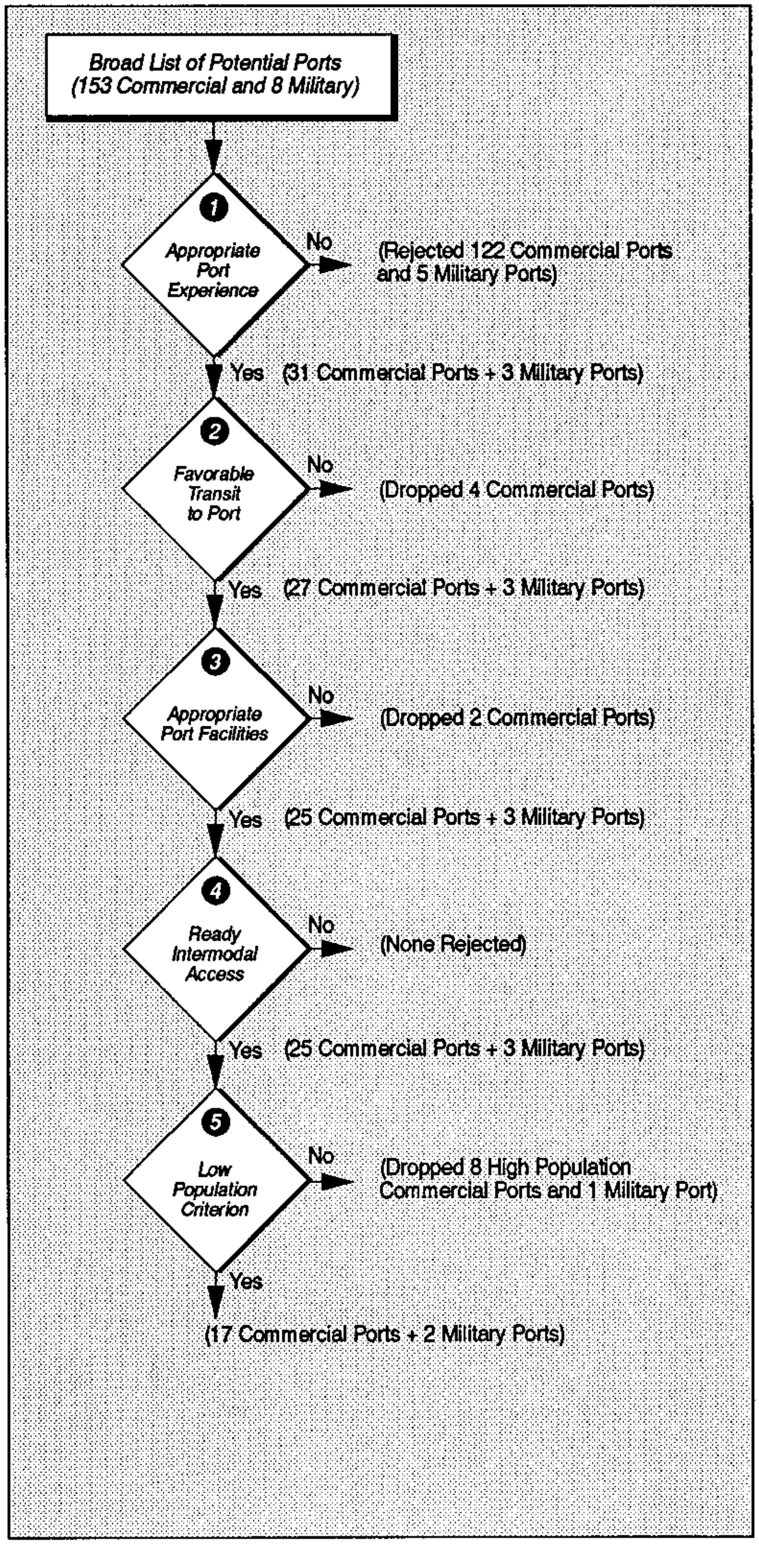


Figure D-18 Screening Ports for DOE "Lowest Human Population" Criteria

D.1.9.6 Desirable Port Attributes

As discussed in Section D.1.9, there are a number of desirable attributes that were not important enough individually to reject an otherwise acceptable port, but have been collectively used to select proposed ports from the list of ports found acceptable under the other DOE criteria. As an element of desirable attributes, DOE examined the likely usefulness of ports for foreign research reactor spent nuclear fuel shipments to any of the five DOE management sites.

The term "usefulness," as used here, is a relative term wherein the relative numbers of scheduled shipping lines and the types of ships that service each port and the countries served by those lines, are compared for two or more otherwise acceptable ports for purposes of selecting the best of that group. This helped to select the ports most likely to be useful in relation to this EIS. This information is shown in Table D-6.

In using these factors, the Ports of Newport News, Norfolk, and Portsmouth, VA, are examined as a single port: Hampton Roads, VA. Table D-6 shows the results of the evaluation of the low population ports for usefulness. The limited usefulness of a port for truck or rail access and service to the potential foreign research reactor spent nuclear fuel management sites eliminated the Ports of Eddystone, PA, Miami and Port Everglades, FL, Wilmington, DE, and Seattle, WA from further consideration.

DOE also identified the most desirable attributes of the remaining ports, such as terminals that do not have conflicting activities nearby (e.g., cruise ship lines, large tourist populations, large petroleum or petrochemical facilities, etc.), and are well-separated from high density populations, have secure short-term storage for contingencies, and have adequate emergency preparedness.

Absence of Conflicting Activities in Port Facilities

While it is the long-held position of DOE, Department of Transportation, and the NRC (NRC, 1977) that spent nuclear fuel shipped in certified Type B casks is well-protected from possible damage due to accidental cask drops, transportation fires, or immersion in water, DOE also concluded that the small public risks associated with such activities could be reduced further if the port in question also had no potentially dangerous, unavoidable conflicting activities, such as regularly scheduled transport of explosive or flammable cargoes, no petroleum carriers or storage facilities in the immediate vicinity of foreign research reactor spent nuclear fuel carrying vessels, and no large numbers of tourists in the immediate area of the terminal who could be unacceptably impacted by a severe accident (good planning and scheduling for arrival of foreign research reactor spent nuclear fuel carrying vessels could mitigate many potential conflicts). The absence of conflicting activities at potential commercial ports was considered in the final port selection to the maximum extent practicable.

A similar important factor for military ports is whether there is adequate separation of the potential port facilities to be used for receipt of the spent nuclear fuel from other hazardous activities (e.g., loading munitions). An exception would be military facilities that were designed and constructed to mitigate the potential impacts of explosions or fires at other piers. Two examples of such facilities which were accepted under the other DOE criteria are MOTSU, NC and the NWS Concord, CA, where such activities are routinely carried out with a high degree of safety. In addition, such conflicts can be avoided by scheduling foreign research reactor spent nuclear fuel shipments at times when no explosives are present at piers.

Table D-6 Relative Usefulness of Low Population Ports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Ports-of-Entry	*Relative Usefulness by Storage Site*					*Relative Usefulness by Foreign Research Reactor Spent Nuclear Fuel Shippers*	
	SRS	*ORR*	*INEL*	*HS*	*NTS*	*Conventional Carriers*	*Charters*
Commercial							
East Coast							
Charleston, SC (Wando Terminal)	T&R	T&R	T&R	T&R	T&R	Europe, Far East, Japan, Australia	Yes
Eddystone, PA	R	No	No	No	No	Central/South America	Yes
Fernandina Beach, FL	T&R	T&R	T&R	T&R	T&R	South/Central America, Mediterranean (monthly)	Yes
Hampton Roads, VA	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Jacksonville, FL	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Miami, FL	T&R	No	No	No	No	Central/South America, Mediterranean, Mexico, Far East	Yes
Port Everglades, FL	T&R	T&R	No	No	No	South American, Northern Europe, Mediterranean, Mideast, Scandinavia	Yes
Savannah, GA	T&R	T&R	T&R	T&R	T&R	Most of the world	Yes
Wilmington, DE	T&R	T&R	T	T	T	Central/South America	Yes
Wilmington, NC	T&R	T&R	T&R	T&R	T&R	Northern Europe, Mediterranean, Mideast, East and South Africa, South America, Far East, Australia	Yes
Gulf Coast							
Galveston, TX	T&R	T&R	T&R	T&R	T&R	Northern Europe, Mediterranean, Mexico, South America, Central America	Yes
Gulfport, MS	T&R	T&R	T&R	T&R	T&R	Northern Europe, Central/South America	Yes
West Coast							
Portland, OR	T&R	T&R	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Seattle, WA	No	T	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Tacoma, WA	T	T	T&R	T&R	T&R	Most of the Pacific Rim, Mediterranean	Yes
Military							
MOTSU (NC)	T&R	T&R	T&R	T&R	T&R	None	Yes
NWS Concord (CA)	No	T&R	T&R	T&R	T&R	None	Yes

SRS = Savannah River Site, ORR = Oak Ridge Reservation, INEL = Idaho National Engineering Laboratory, HS = Hanford Site, NTS = Nevada Test Site; T = truck, R = rail

Emergency Response Capabilities

The U.S. Merchant Marine Academy Workshop identified the importance of a risk management staff and emergency response capabilities (including response plans and training of operating personnel) in determining the acceptability of ports for receipt and handling of foreign research reactor spent nuclear fuel. DOE focused on identification of ports that have current emergency response plans and personnel appropriately trained to respond to a port emergency to protect workers and the public from avoidable risks (however small). Since few ports have detailed response plans for radiological emergencies involving spent nuclear fuel, DOE determined that such shortcomings do not prevent consideration of such ports provided the ports have in place appropriate response plans and training for hazardous cargo accidents, since many of the features are the same (e.g., identification of decisionmakers, first responders, and support personnel to mitigate impacts of fires, etc.). In addition, for ports that have no specific response plans for spent nuclear fuel accidents in port, DOE could provide assistance in the development of radiological emergency response plans (in addition to existing hazardous cargo emergency response capability) and training at such ports in the event they were ultimately selected for foreign research reactor spent nuclear fuel shipments. Thus, appropriate plans and training would likely be in place prior to actual receipt of any such shipments. Ports having current emergency response capabilities were considered more desirable than those that do not.

Spent Nuclear Fuel Handling Experience

The National Defense Authorization Act would also require, "to the maximum extent practicable," that the ports selected for receipt of foreign research reactor spent nuclear fuel have spent nuclear fuel handling experience. At the present time, there are only a few ports in the United States with relatively recent experience handling either spent nuclear fuel or high-level radioactivity in Type B casks. As a result, this criterion, while desirable, unnecessarily restricts considerations to an unacceptably small group of potential ports, and strictly applied, could preclude shipments of spent nuclear fuel from some of the countries being considered under this EIS except by chartered ship. However, because all containerized cargoes are handled in the same manner as the containerized spent nuclear fuel would be handled, DOE concluded that current experience (especially any involving routine handling of potentially hazardous cargoes, or other radioactive cargoes in Type B casks) is much more important for public safety than foreign research reactor spent nuclear fuel handling experience in years past. This is especially true since the trained longshoremen are likely to have changed jobs, ports, or retired during the several years between the last shipments of spent nuclear fuel and the potential onset of future shipments under this EIS.

In addition, ports that have satisfied the "appropriate experience" and "port facilities" criteria are expected to be fully capable of currently handling spent nuclear fuel containers, and would gain experience as the program progressed.

Environmental Concerns Near Ports

Marine areas, immediately surrounding most of the ports considered in this selection process, tend to be severely impacted as a result of necessary periodic dredging or construction of new port facilities, including turning basins, high volumes of marine traffic, and routine port activities. As a result, ports generally are no longer environmentally sensitive areas within the context of NEPA. However, consistent with U.S. Merchant Marine Academy Workshop recommendations and in response to public comments, DOE decided that when special protected or sensitive areas were identified nearby the terminal(s) being considered, these areas would be identified in the EIS and used for final port identification as appropriate. No serious issues have been identified in the immediate vicinity of any ports selected under the DOE low population criterion review, with the possible exception of the NWS Concord, CA and Fernandina Beach, FL.

Environmental Concerns from Severe Natural Phenomena

Other factors that were considered desirable attributes for ports include average or lower risks from severe weather (e.g., extremely high winds, hurricanes, etc.) or other natural phenomena (e.g., seiches, earthquakes, volcanism, etc.). These attributes are not expected to be of great significance in practice, since the time involved with potential receipt and transhipment of containerized spent nuclear fuel represents such an extremely short period of risk (typically less than 24 hours), that the probability of severe natural phenomena impacting foreign research reactor spent nuclear fuel shipments is vanishingly small. Further, some natural events, such as hurricanes, can often be avoided. However, these characteristics were examined in conducting the port evaluation.

Separation of Port Facilities from Urban Populations

The following desirable characteristics are examined:

- Terminals used for spent nuclear fuel shipments should be physically separated from densely populated city centers (by several kilometers if possible) to help ensure that the general public would be unlikely to be exposed to significant radiation doses from either incident-free transport or accidents within the port (e.g., cask drops, fires, or truck or rail accidents, etc.).
- Transport of spent nuclear fuel through large, densely-populated, congested areas around the port should be avoided where practical.

These geographic/demographic characteristics, while not explicitly addressed in the evaluation of "lowest human populations" for ports, are implicitly included in the 16 km (10 mi) radius populations used for screening ports. While absence of these characteristics would not necessarily eliminate the use of such ports under this EIS, DOE reviewed these ports to determine if there were terminals or piers within the port that provided these characteristics. In many cases, development of new port facilities in recent years has resulted in specific terminals and/or piers that meet all of the required criteria (USMMA, 1994, and NDAA, 1993), and that also have most or all of the additional desirable characteristics (e.g., the Wando Terminal in Charleston, SC, the Blount Island Terminal in Jacksonville, FL, or Terminal T6 in Portland, OR).

Absence of Local Restrictions on Receipt and Handling of Spent Nuclear Fuel

Another desirable port factor recommended by the U.S. Merchant Marine Academy Workshop is the absence of local regulatory restrictions on receipt and handling of spent nuclear fuel. It is well established that local restrictions on international or interstate commerce are void under the U.S. Constitution, and similar challenges have been rejected by the Federal courts. For example, the Port of Oakland, CA indicated that a citizen's legislative initiative in 1987 led to a ban on the handling and transport of foreign research reactor spent nuclear fuel through the port. Although Oakland's ban was invalidated by the Federal District Court, the Port Authority has maintained some control over radioactive shipments through the port through its permitting system (Adams, 1993). Nevertheless, although claiming to be a "nuclear free zone," the port continues to allow permitted shipments of certain radioactive materials, handling approximately 500 metric tons (551 tons) of radioactive shipments between January and June 1994 (Adams, 1994).

Further, if DOE were to avoid selection of ports with restrictions by local ordinances, every port wishing to close its doors to receipt of spent nuclear fuel (or any other type of cargo) would simply promulgate an ordinance. Therefore, the EIS will only identify existing local restrictions (formal or informal) in section D.2 for consideration by decisionmakers, and this criterion will have no immediate impact on determination of the acceptability of ports within this EIS.

Secure Short-Term Storage

Although the National Defense Authorization Act requires, to the extent practicable, expeditious movement of casks from a port, the presence of regular guards, fences, and lighted areas that provide security at all times is a desirable attribute. Such additional features provide assurance of safe segregation and short-term storage of foreign research reactor spent nuclear fuel shipments away from workers and the public in the event of unexpected local occurrences, such as snow or ice storms, traffic congestion, and other events beyond the control of spent nuclear fuel shippers.

To best comply with this attribute, the storage area should be one designated for the storage of hazardous materials (referred to as a facility of particular hazard). Such designations are normally simple processes which result in U.S. Coast Guard approval following a request by the terminal operator. While all the military ports are designated as "facilities of particular hazard," some commercial facilities may only request periodic designations for specific incoming or outgoing cargoes (e.g., the Port of Tacoma, WA periodically designates Terminal 7B for occasional shipments of potentially explosive ammonium nitrate). Table D-7 shows which commercial ports have traditionally had secure storage areas for hazardous cargoes, and DOE has assumed such storage would be available in the future for receipt and short term storage of foreign research reactor spent nuclear fuel. (More detailed information on "facilities of particular hazard" may be found in section D.4.3).

D.1.10 Application of the Desirable Port Attributes in Port Selection

As a result of the evaluation of desirable attributes, two additional ports, Fernandina Beach, FL, and Gulfport, MS, were removed from the potential ports of entry list (Table D-7). The port of Fernandina Beach, FL, is not well-separated from the urban population surrounding the port, and the population is expected to substantially grow by about 82 percent by the year 2010 (see Attachment D2). Also, entry to the port requires ship passage through a State sea manatee (an endangered species) preserve. The Port of Gulfport, MS, does not currently have a well-secured area designated for the storage of foreign research reactor spent nuclear fuel, and it is unlikely it ever will due to casino operations. There is a former cruise ship terminal at the East Pier, which is slated for new casino development, a floating casino located in the port and two new casinos on the West Pier. In addition, the port is not well-separated from surrounding urban population.

Conclusion

As a result of the evaluation, ten ports remained as the final list of ports acceptable for the potential receipt, handling, and transhipment of foreign research reactor spent nuclear fuel. These ten ports [Charleston, SC; Galveston, TX; Hampton Roads (includes terminals in Newport News, Norfolk, and Portsmouth), VA; Jacksonville, FL; MOTSU, NC; NWS Concord, CA; Portland, OR; Savannah, GA; Tacoma, WA, and Wilmington, NC] represent the final list of ports considered for the receipt of foreign research reactor spent nuclear fuel.

D.2 Detailed Information on Potential Ports of Entry

This section of Appendix D provides detailed information that served as the bases for identifying the candidate ports addressed in Section D.1. For convenience, the port details are divided into two categories: (1) the DOE candidate ports of entry that met the criteria developed for port identification in Section D.1, and (2) the remainder of the ports that fully or marginally satisfied the first criterion for appropriate port experience. Within each of the categories, the ports are arranged in alphabetical order. The location of the ports is shown in Figure D-1.

Table D-7 Use of Desirable Attributes for Selecting Final "Low Population" Ports for Foreign Research Reactor Spent Nuclear Fuel Shipments

Ports-of-Entry	*Free of Conflicting Uses at Port Facilities*	*Emergency Preparedness*	*Short-Term Secure Storage*	*Free of Environmental Concerns*	*Free of Severe Natural Phenomena*	*Terminal Well-Separated from High Density Populations*
			Commercial			
East Coast						
Charleston, SC	Yes (Wando)	Yes	Yes	Yes	E, H	Yes (Wando)
Fernandina Beach, FL	T	Yes	No	Some (Manatee)	H	No
Hampton Roads, VA	Yes	Yes	Yes[b]	Yes	Yes	Yes (Newport News, VA)
Jacksonville, FL	Yes (Blount Island)	Yes	Yes	Yes	H	Yes (Blount Island)
Savannah, GA	P	Yes	Yes[b]	Yes	E, H	Yes (Container Port)
Wilmington, NC	P, Some Ex[a]	Yes	Yes[b]	Yes	H	Yes
Gulf Coast						
Galveston, TX	Some P, T, Ex[a]	Yes	Yes[b]	Yes	H	No
Gulfport, MS	T[c]	Yes	No	Yes	H	No
West Coast						
Portland, OR	Yes	Yes	Yes[b]	Yes	E, V	Yes (T6)
Tacoma, WA	Yes, some Ex[a]	Yes	Yes	Yes	E, V	Yes
			Military			
MOTSU (NC)	Ex[a]	Yes	Yes	Yes	H	Yes
NWS Concord (CA)	Ex[a]	Yes	Yes	Some (wetlands and Tule elk)	E	Yes

Ex= explosives, T = tourism, P = petroleum handling/storage facilities, H = hurricanes/tropical storms, V = volcanoes, E = earthquakes

[a]*Separation of piers and scheduling of spent nuclear fuel and explosive shipments on different days makes consideration of these ports appropriate*

[b]*No currently designated facilities of particular hazard at preferred terminal(s)*

[c]*Extensive casino development within 1,000 feet*

D.2.1 Detailed Information on Candidate Ports of Entry

D.2.1.1 Charleston, SC (Includes the Naval Weapons Station Terminal and the Wando Terminal)

Charleston is the largest port city in South Carolina, and the greater Charleston area is one of the major seaports on the East Coast of the United States. The city of Charleston itself is located at the confluence of the Cooper and Ashley Rivers, approximately 11 km (7 mi) from the entrance from the sea. The principal wharves are along the west bank of the Cooper River except for the Wando Terminal which is along the east bank of the Wando River near Mount Pleasant, about 20 km (11 mi) from the Atlantic Ocean. The city is the center of a rich agricultural district for which it is the distribution point. The entrance to the harbor is maintained by a Federal project providing a channel depth of 10.7 m (35 ft) over the bar, through the entrance and into the major reaches of the Cooper River. The harbor is easy to access in day or night in clear weather, and is one of the best harbors of refuge on the South Atlantic coast (DOC, 1993d). The maps of the port are shown in Figures D-19 (Naval Weapons Station, Charleston) and D-20 (Wando Terminal).

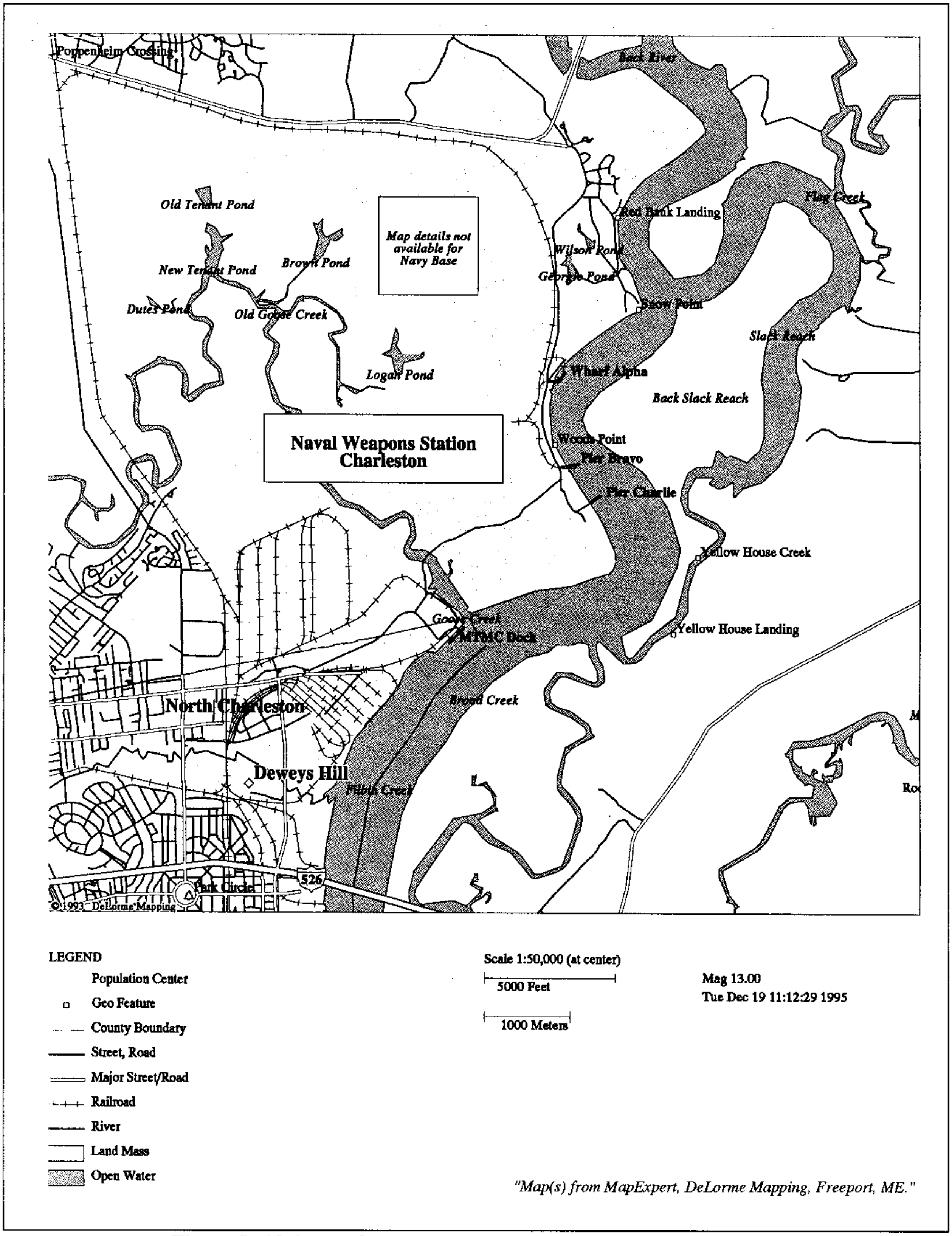


Figure D-19 Map of the Naval Weapons Station, Charleston, SC

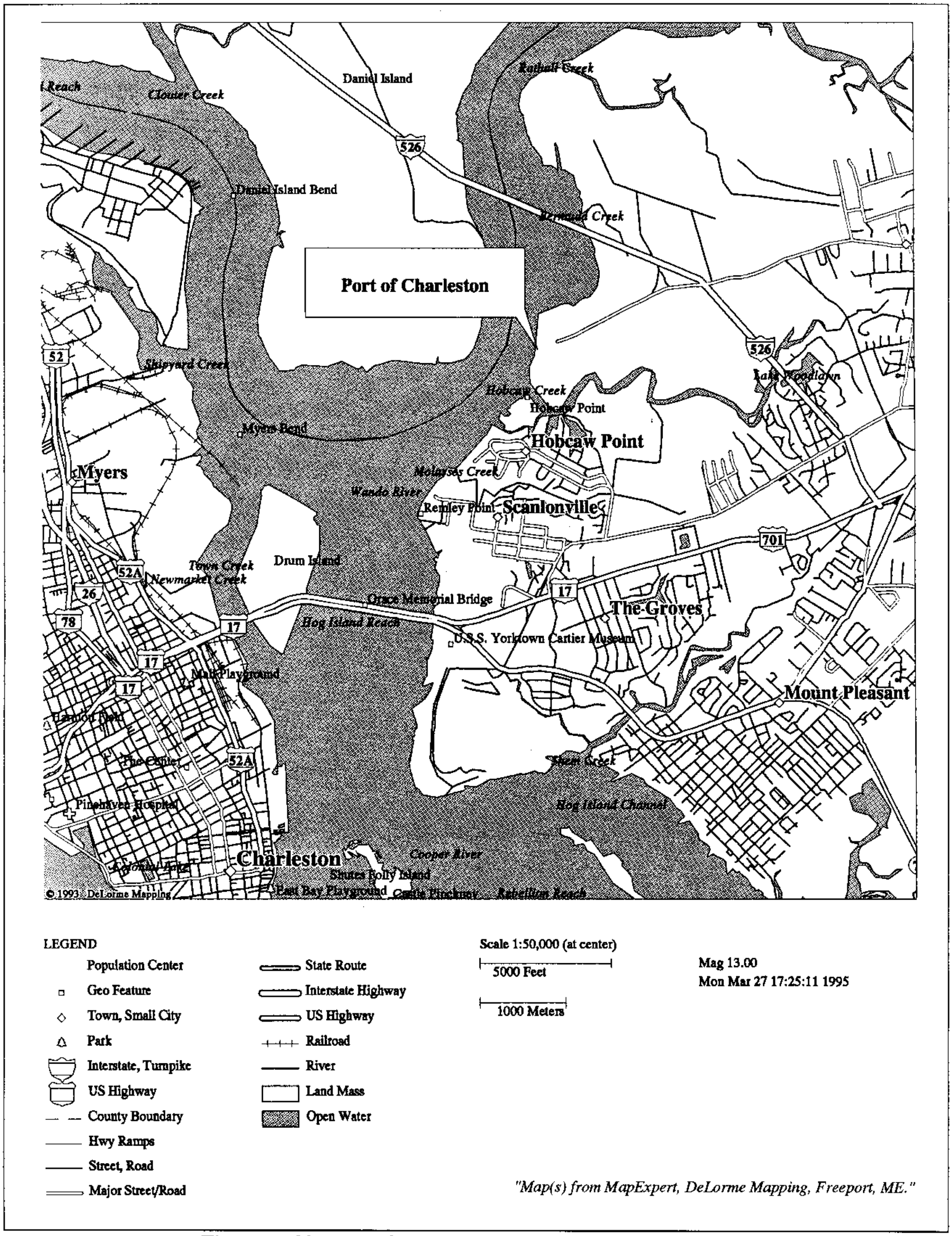


Figure D-20 Map of the Wando Terminal, Charleston, SC

However, the areas to the east and southeast of the port entrance are used extensively by the U.S. Navy and other military services for training exercises which may result in occasional restrictions. Under unfavorable weather conditions current velocities in some areas have been reported as high as 2.1 meters-per-sec (4 knots) (DOC, 1993d). All of the port terminals have 12.2 m (40 ft) of water alongside at mean low water. The port is serviced by many of the world's largest container shipping lines (a total of 56), that handled 807,106 standard 20-ft container equivalents in 1991 (AAPA, 1993; FHI, 1993a). These lines provide service between Europe, the Far East, Japan, Australia and other countries (Jane's, 1992).

The South Carolina State Ports Authority owns and operates four large general cargo and container terminals within the greater Charleston area. The City of Charleston hosts two facilities (Union Pier Terminal and Columbus Street Intermodal Terminal) that were eliminated from consideration because they are not well separated from dense urban populations, and are within the city limits and subject to potential restrictions on receipt and handling of spent nuclear fuel (Jane's, 1992; AAPA, 1993).

The North Charleston Terminal is a container terminal located about 16 km (10 mi) upstream from the city of Charleston. This facility was considered to be inferior to the Wando Terminal because it requires additional transport up a heavily trafficked and more confining channel (only about 120 m (or 400 ft) wide in many reaches) on the upper Cooper River, with ships required to pass below an additional bridge (I-526) over the river (in comparison to Wando Terminal). Further, superior facilities and better separation from populated areas are found at the Wando Terminal discussed below.

In the Draft version of this EIS, only the Wando Terminal was addressed in detail. Public commentors from the Charleston area and other candidate port areas suggested that DOE further consider military ports. Since the Draft EIS was published, the Record of Decision for the SNF&INEL Final EIS (DOE, 1995) directs all aluminum-based spent nuclear fuel to the Savannah River Site. Because of the public requests and the relative proximity of the Savannah River Site to the greater Charleston area, the NWS Charleston has been added as a candidate port of entry, and detailed information is provided in the following section.

Other Pertinent Information: The City of Charleston has a city ordinance restricting the transport of spent nuclear fuel through the city. According to information gathered, the ordinance does not preclude shipment, but requires a permit and approval from the city. The Sandia National Laboratories Radioactive Materials Postnotification Database indicates that the port has not received any spent nuclear fuel since the database was initiated in October 1984 (SNL, 1994), and the NRC has no record of foreign research reactor spent nuclear fuel shipments since 1979, when NRC began approving spent nuclear fuel shipments (NRC, 1993). From discussions with senior port officials, it was determined that Wando Terminal would handle spent nuclear fuel shipments provided they had the approval of the U.S. Coast Guard Captain of the port and the Charleston Fire Department (Moise et al., 1993). Use of City terminals, rather than the Wando Terminal, has the potential for delays in the receipt and transhipment of foreign research reactor spent nuclear fuel, which could result in failing to move the foreign research reactor spent nuclear fuel from the port of entry to the management site "expeditiously." The NWS Charleston is capable of handling spent nuclear fuel shipments provided that the NWS Charleston receives appropriate program "Waivers". A program waiver would have to be issued by the Chief of Naval Operations to allow NWS Charleston facilities to be used to handle spent nuclear fuel shipments. Event waivers would have to be issued by the NWS Charleston Commanding Officer to allow each shipment to be handled. Event waivers are routine procedures used by the NWS Charleston Commanding Officer to place restrictions on conflicting activities, such as ammunition handling (Stark, 1995).

The South Carolina State Ports Authority Port Police are part of an emergency response team comprised of the local fire departments, Coast Guard, and private hazardous materials response organizations. The Ports Authority provides operating personnel basic hazardous materials training. Dock workers are trained in hazardous materials placard recognition and other basic information by the port's stevedores. Security is provided by perimeter fencing with controlled access and the South Carolina State Ports Authority Police Force, which maintains 24-hour manned access booths, patrols, and surveillance. All container terminals have secure, open and/or covered storage space for temporary storage of spent nuclear fuel if necessary (Moise et al., 1993).

The Wando Terminal is located several kilometers northeast of downtown Charleston in a relatively low population area with good access to interstate highways. Aside from general environmental concern for the wetlands around the port, there are no known special sanctuaries or habitats of concern although the port is subject to severe hurricanes (with high water) and tropical storms. It was the site of the largest earthquake (Modified Mercalli Intensity X) in the Eastern United States in recorded history, on August 31, 1886 (Bolt, 1978). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Charleston, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a moderate seismic zone with an acceleration of 0.15 g.

There are several tanker terminals and petroleum storage depots along the west bank of the Cooper River downstream of the North Charleston Container Terminal (which is also immediately adjacent the Naval Weapons Station off Goose Creek). However, there do not appear to be any conflicting cargoes or activities at the Wando, Columbus, or Union Pier Terminals. The port officials contacted indicated that they believe that radioactive shipments have been made through the port in the past, but they were not sure if spent nuclear fuel had been handled (Moise et al., 1993).

Environmental Conditions

The State of South Carolina has given the lower portion of the Wando River two different water quality classifications. The water is classified as SFH or SA. SFH waters are shellfish harvesting waters and SA waters are suitable for primary and secondary recreation and for other water uses requiring lower quality. According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for James Island, SC, the Wando Terminal and the NWS Charleston are located in a mid-salinity estuarine habitat (generally 5 to 16.5 parts per thousand). The Charleston Harbor, which is traversed on the way to either terminal, is located in a high-salinity estuarine habitat (generally 16.5 to 30 parts per thousand) (FWS, 1980b).

The State of South Carolina has also classified the water quality of the portion of the Cooper River above the confluence with the Ashley River as SB (SB waters are tidal saltwaters suitable for secondary contact recreation, crabbing, and fishing, except the harvesting of clams, mussels, or oysters for market purposes and human consumption). The waters of Goose Creek, upstream of the confluence with the Cooper River to the dam at the Charleston Waterworks, are also Class SB (Department of the Navy, 1994).

The lower Wando River, the Charleston Harbor, and the NWS area support a large number of aquatic and terrestrial species. According to the South Carolina Heritage Trust, no rare, threatened, or endangered species or communities have been recorded in the area near the Wando Terminal (McBee, 1994). State or Federally protected endangered or threatened aquatic species in the vicinity of the Charleston Harbor include the Shortnose sturgeon, Atlantic sturgeon, and the American shad. Bachman's warbler is a Federally protected bird species also found in the vicinity (FWS, 1980b).

In addition, the U.S. Fish and Wildlife Service reports that several protected marine species are known to occur in Charleston County (Banks, 1994). These are the west indian manatee (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), and the green sea turtle (threatened). Protected bird species include the arctic peregrine falcon (threatened), bald eagle (endangered), wood stork (endangered), red-cockaded woodpecker (endangered), and the piping plover (threatened).

In recent years, two pairs of bald eagles (*Haliaeetus leucocephalus*) nested on the NWS Charleston. One nest was located north of Foster Creek near the POMFLANT storage and activity area, over four miles north of Wharf Alpha. The other was located on the golf course west of Pier X. Their nest tree was destroyed by Hurricane Hugo and the pair have relocated to Big Island, located north of Foster Creek (Department of the Navy, 1994).

Prior to Hurricane Hugo, 12 colonies of red-cockaded woodpeckers, *Picoides borealis*, were known at the NWS Charleston. The red-cockaded woodpecker requires mature pines old enough to be susceptible to red heart disease, which makes the wood soft enough for these small woodpeckers to create a cavity. Such trees are generally at least 70 to 75 years old. Nearly all trees this age were destroyed by Hurricane Hugo, so it is unlikely that the colony could reestablish at this site in the near future (Department of the Navy, 1994).

Wetlands are plentiful on and adjacent to the NWS Charleston. Three basic habitat types found within the area's wetland ecosystem are forested wetlands, nonforested wetlands, and open water. The station's wetland habitats had previously been identified according to the National Wetland Inventory classification system. Within this classification system, the station's wetlands had been placed in four major categories: estuarine, palustrine, lacustrine, and riverine. Based upon that classification, the station contained 1,356 acres of estuarine, 1,730 acres of palustrine and 131 acres of lacustrine wetlands (Department of the Navy, 1994). Field investigations have been performed at the NWS, and no rare or endangered plants or animals were observed (Department of the Navy, 1990 and 1994).

Wetlands at the station contain potential habitat for the flatwoods salamander (*Ambystoma cingulatum*), which is pending placement on the threatened or endangered species list. However, two spot checks of the area by the NWS Charleston environmental personnel have failed to locate either adults or larvae. It has been indicated that a detailed study may be performed of the area in the future (Department of the Navy, 1994).

Climatic Conditions

In general, the elevation of the area ranges from sea level to approximately 6 m (20 ft) on the peninsula. The climate of this region is temperate, primarily due to its close proximity to the Atlantic Ocean. The prevailing winds are generally northerly in the fall and winter months, becoming more southerly during the summer months. This type circulation serves to "warm" the region during winter and "cool" it during the summer. Summer is the rainy season in Charleston, with the city receiving 41 percent of the annual total rainfall during these months. Except for the occasional tropical storm or hurricane, the majority of this rain occurs during afternoon and evening thunderstorms. The late summer and early fall brings the highest probability of tropical storm activity to the Charleston, SC area. The fall season is a transitional period, where temperature extremes are rare and sunshine is abundant. The winters in this area are mild with periods of rain. However, in contrast to the summer, the winter rains tend to be steady and uniform, and last for several days. The most unstable period in this region is spring when the confluence of warm moist tropical air and cool dry continental air increase the occurrence of severe weather in this region. The average earliest freeze in this area is in early December and the average last frost is in late February (NOAA, 1992c).

D.2.1.1.1 Naval Weapons Station - Charleston

The NWS Charleston is located on the west bank of the Cooper River, north of the city of North Charleston in southeastern Berkeley County, South Carolina. The station occupies about 7080 hectares (17,500 acres) along a 14-km (9-mi) stretch of the Cooper River, starting about 30 km (19 mi) from the Atlantic Ocean. The primary missions of the NWS Charleston are to provide material support for assigned weapons and weapon systems, to provide housing and community support facilities for personnel assigned to the Charleston area, and to do additional tasks such as home porting and logistics support for ammunition ships, and other fleet and shore activities dealing with weapons. Major tenant activities on the station include the new Army Strategic Mobility Logistics Base, a Propulsion Training Facility and the Military Traffic Management Command, an Army organization (Department of the Navy, 1990 and 1994). The Army Strategic Mobility Logistics Base is being constructed on the formerly Polaris Missile Facility Atlantic site (Lewis, 1995).

In selecting a port this far from the open ocean, DOE considered the navigation safety through the Charleston Harbor and up the Cooper River. As previously described, the harbor experiences a significant amount of deep draft traffic, and is accustomed to managing ship traffic in several 10.7 m (35 ft) deep channels. The Navy maintains a 10.7-m (35-ft) deep channel up the Cooper River to all relevant piers and wharfs of the NWS. The channel is as narrow as 120 m (400 ft) in some areas and extends to the edge of piers that may be in use for handling ammunition, petroleum products or other hazardous cargo (DOC, 1993d). However, with proper management of the harbor by the U.S. Coast Guard and pilots, and the proper planning by the U.S. Navy, additional assurance of a safe transit can be provided.

The Charleston Harbor Navigational Guidelines (DOC, 1993d) identifies areas of particular concern in the Cooper River area and provides guidelines for navigation, overtaking and passing in these areas. The Charleston Branch Pilots Association procedures (Commissioners of the Pilotage, 1995) require strict adherence to these guidelines for deep draft vessels and vessels carrying hazardous materials. The Coast Guard Captain of the port has broad regulatory authority over all port activities, and procedurally delegates control of vessel movements to the Charleston Branch Pilots Association (Bennett, 1995). This authority includes activities ranging from minor additional traffic restrictions on passing to establishing and enforcing a moving safety zone around a ship traversing the harbor area. A moving safety zone requires advance notification in the *Federal Register*, but is routinely done in areas in which hazardous materials are carried (USCG, 1994c). The Coast Guard and the South Carolina State Police have conducted exercises on moving a ship up the Cooper River under the threat of terrorist activities (Millar, 1995).

The U.S. Navy is the only significant user of the Cooper River north of the North Charleston facilities, and is a major user of the Cooper River. Cooper River transits can be planned by the Navy so that they would not conflict with other Naval activities and ship movements. The Commanding Officer of the NWS Charleston would determine which facility is most appropriate for each shipment based on the characteristics of the vessel carrying the fuel, the planned mode of overland transportation and any conflicting activities at the NWS Charleston. The NWS Charleston has four facilities that can handle spent nuclear fuel. The northern facility is Wharf Alpha, which is more than 12 m (40 ft) wide and has about 300 m (1000 ft) of useful berthing area dredged to 12.3 m (40 ft) deep. Wharf Alpha was previously used to service Polaris missile-carrying nuclear submarines, including removal and replacement of nuclear missiles. This function is no longer necessary, and submarines do not regularly visit the NWS Charleston.

Wharf Alpha is currently being used to service and load U.S. Navy ammunition ships. With the construction of the Army Strategic Mobility Logistics Base, the Army plans to upgrade in 1998 the rail lines serving Wharf Alpha as well as expanding the wharf itself. The wharf has a mobile crane, and is served by truck and rail loops, meaning that either trucks or trains can drive directly onto the wharf, load, and exit without turning around or reversing direction. Pier Bravo, located about 1 km (3281 ft) south of Wharf Alpha, is also used for ammunition handling. Pier Bravo protrudes 300 m (983 ft) into the Cooper River, with 214 m (703 ft) on the south side and 166 m (545 ft) on the north side of useful berthing in 11.9 m (39 ft) deep waters. The pier has a mobile crane that can off-load spent fuel casks directly onto trucks or trains parked on the pier.

Pier Charlie, located about 1000 m (3281 ft) south of Pier Bravo, has been used for berthing nuclear submarines and a tender. It is not in regular use, but is maintained as a backup for Wharf Alpha and Pier Bravo. A portable crane, which is capable under ideal conditions of lifting 36,280 kg (40 tons), can be moved to Pier Charlie and used to off-load casks onto trucks. The Military Traffic Management Command dock is located about 3 km (9840 ft) down the river from Pier Charlie. This facility is in regular use for roll on/roll off military cargo. The Military Traffic Management Command dock is a safe distance from any weapons handling operations at Wharf Alpha or Pier Bravo. However, the same portable crane that could be used at Pier Charlie would be used at the Military Traffic Management Command dock.

The portable cranes available on the NWS Charleston may not be able to off load some larger casks, especially if they have to extend horizontally over the ship. The DOE and the NWS Charleston would plan to use shipboard cranes or rented cranes, or schedule these certain shipments to dock at Wharf Alpha or Pier Bravo. Commercial 82 metric ton (90 ton) and 118 metric ton (130 ton) cranes are available in the greater Charleston, SC area (Silver, 1995). Additionally, Pier Charlie and the Military Traffic Management Command dock are not directly served by rail. However, several rail heads on the NWS are in secure and isolated locations that can be used to load the fuel from trucks to trains.

The NWS Charleston is a fenced facility with several guarded gates and a 24 hour security force. Additional guard facilities and temporary barricades are used to keep unnecessary personnel away from ammunition handling, and could be used for this program. The on-base emergency facilities are appropriate for fire and rescue response to ammunition handling and other potential accidents. Additionally, the Propulsion Training Facility includes two operating nuclear reactors. The staff is adequately trained and equipped to make initial response and assessment of any accident with the potential for radioactive release or radiation exposure.

The NWS Charleston facility is capable of supporting the implementation of a policy to accept foreign research reactor spent nuclear fuel. The facility is experienced in handling nuclear and hazardous cargo, can be safely reached from open ocean, has adequate facilities, ready access to truck and rail transportation, and low human populations. Desirable attributes include excellent emergency response capabilities, acceptable environmental concerns, moderate concerns from severe natural phenomena, separation from urban population, no local restrictions and secure short term storage. The risk associated with the conflicting uses can be mitigated by rigorous compliance with Naval operation procedures.

The 1990 population within 16 km (10 mi) of the Wharf Alpha was 209,188. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 46,200; Oak Ridge Reservation, 108,000; Idaho National Engineering Laboratory, 498,000; Hanford Site, 550,000; and Nevada Test Site, 540,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: Savannah River Site, 301 km (188 mi); Oak Ridge Reservation, 644 km (402 mi); Idaho National Engineering Laboratory, 3,910 km (2,441 mi); Hanford Site, 4,580 km (2,858 mi); and Nevada Test Site, 3,930 km (2,543 mi). Distances along rail routes are slightly longer.

D.2.1.1.2 Wando Terminal

The South Carolina State Port Authority Wando Terminal is an ultra modern marine facility that is the designated hazardous materials terminal for the port ("Facility of Particular Hazard"), and is a superior terminal for receipt of spent nuclear fuel. In addition to being outside the city limits of Charleston and not subject to any potential restrictions on receipt and handling of spent nuclear fuel, it is closest to the Atlantic Ocean, and has outstanding facilities. The terminal has 3 container berths and 67.7 ha (167 acres) of paved container storage yard. It has a 428 m (1,400 ft) by 427 m (1,400 ft) turning basin. It currently has 740 m (2,430 ft) of lineal berthing space, but a fourth berth and 35.2 ha (87 acres) of additional paved storage area is currently under construction. The terminal is 8.1 km (5 mi) from the Mark Clark Expressway (I-526), which by-passes most of the city of Charleston and joins Interstate 26 at North Charleston. Of the four terminals in the Port of Charleston, Wando is the only one without direct rail service, requiring trucking of containers about 15 km (9 mi) to intermodal rail yards serviced by the CSX and Norfolk Southern Railroads. This was not considered a serious problem, since most shipments are anticipated to be carried overland by trucks.

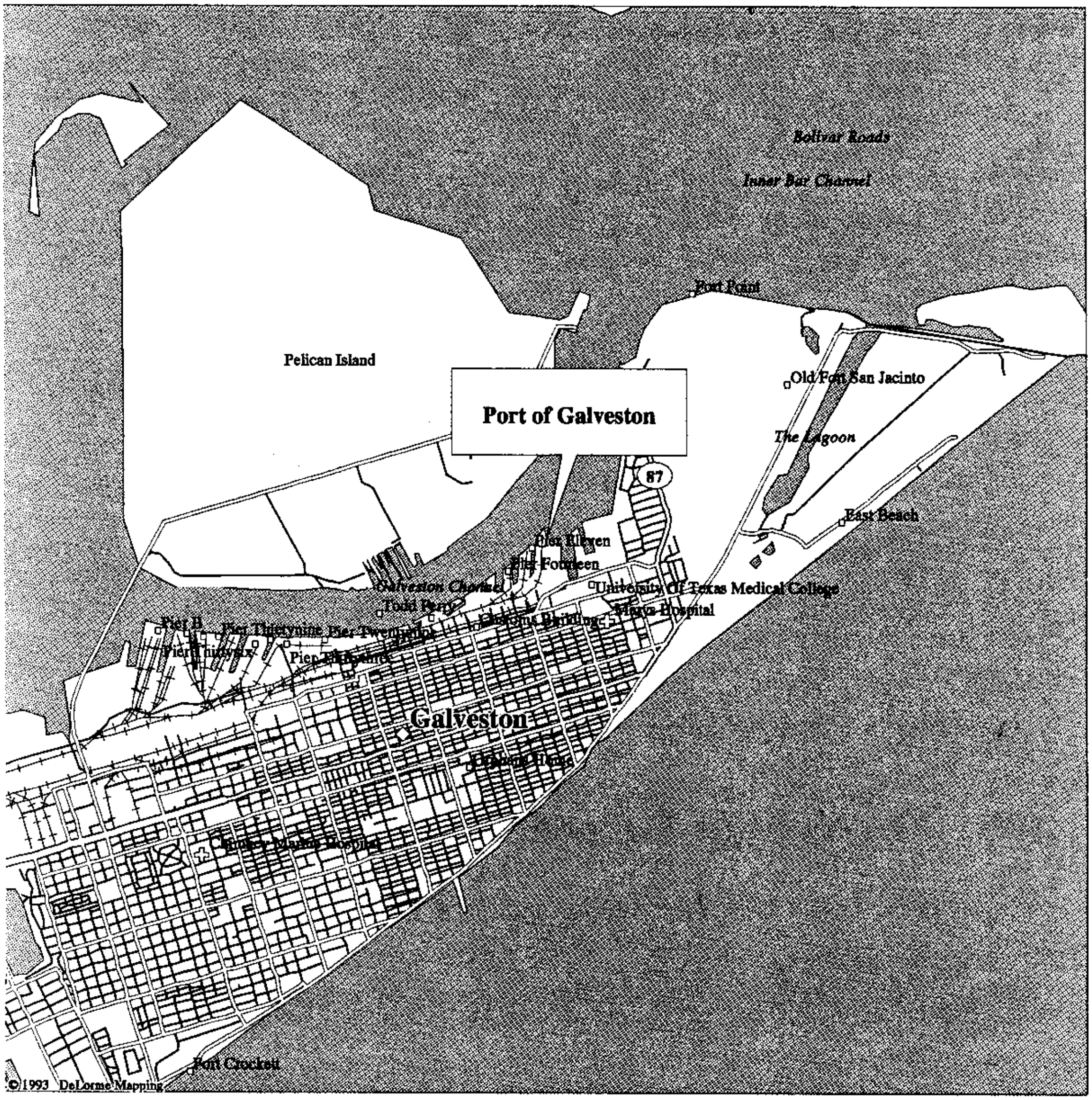
The 1990 population within 16 km (10 mi) of the Wando Terminal was 233,434. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 65,700; Oak Ridge Reservation, 127,000; Idaho National Engineering Laboratory, 518,000; Hanford Site, 569,000; and Nevada Test Site, 559,000. Populations along rail routes to these sites are slightly larger. The distances to the five potential sites on interstate routes are: Savannah River Site, 325 km (203 mi); Oak Ridge Reservation, 668 km (417 mi); Idaho National Engineering Laboratory, 3,935 km (2,456 mi); Hanford Site, 4,600 km (2,873 mi); and Nevada Test Site, 4,098 km (2,558 mi). Distances along rail routes are slightly longer.

D.2.1.2 Galveston, TX

The Port of Galveston is about 16 km (10 mi) from the Gulf of Mexico via the Galveston channel. The City of Galveston, TX, occupies the entire width of the east end of Galveston Island. The shipping wharves are on the north side of the island and the Gulf of Mexico is on the south. The Port of Galveston is located in the heart of the City (DOC, 1992a). A map of the port is shown in Figure D-21.

As stated in the Coast Pilot, the Port of Galveston offers a short route to the sea and, together with the deep and easily navigated channel and excellent port facilities, enables Galveston to handle cargo most expeditiously and economically (DOC, 1992a). A Federal project provides for an entrance channel and an outer bar channel both dredged to 12.8 m (42 ft), thence 12.2 m (40 ft) to Galveston. The Port of Galveston is a multi-terminal port complex located on the northeastern end of Galveston Island, only 15 km (9.3 mi) from the entrance buoy to the open sea. Overall tonnage reported for 1991 was 4,159,233 metric tons (4,584,723 tons), of which approximately 17 percent (703,511 metric tons or 773,862 tons) was containerized freight (over 70,000 20-ft equivalent units). Roughly 77 percent of the tonnage was dry and liquid bulk, much of it grain (AAPA, 1993).

The Port of Galveston is a separate utility of the City of Galveston with its powers established by the City Charter. The Charter provides that all city-owned wharf and terminal properties be set aside and controlled, maintained, and operated by a "Board of Trustees of the Galveston Wharves."



LEGEND

Population Center
State Route
Geo Feature
Large City
Street, Road
Major Street/Road
State Route
Railroad
River
Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 08:46:03 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-21 Map of the Port of Galveston, TX

Principal container lines and the areas they serve include: Lykes Brothers — North Europe, Mediterranean, Mexico and West Coast of South America; Deppe Line — North Europe; Companhia Maritima Nacional — Brazil/Mexico; Compania Chilean Navegacion Interoceanica — South America/Mexico, and Del Monte/Network Shipping — Guatemala/Mexico (Jane's, 1992; AAPA, 1993).

Pier 10 Container Terminal: This is Galveston's principal container handling facility. It is leased to a private operator, Container Terminal of Galveston, Inc., who operates the facility as a public terminal. This facility has two berths, four container cranes, and 19.83 ha (49 acres) of paved storage area. The Port of Galveston owns an additional ten (10) open-dock ship berths and 20 berths with shipside warehouses used for breakbulk and other cargoes. The Container Terminal of Galveston, Inc. has two berths with a total length of 410 m (1,346 ft). Depth alongside the Container Terminal of Galveston, Inc. at mean low water is 12.2 m (40 ft). Crane capacities on Container Terminal of Galveston, Inc's. Pier 10 are three 50.8 metric tons (56 ton) container cranes and one 61.0 metric tons (67 ton) container crane. All cranes are equipped with 40.6 metric tons (45 ton) capacity spreaders.

The Container Terminal of Galveston, Inc. has a controlled all-weather truck entrance and interchange area. The terminal is connected to Interstate Highway 45 on the mainland by the 9.3 km (5.8 mi), four-lane State Highway 87 and two 2.8 km (1.75 mi) causeways that cross the southwest end of Galveston Bay. The island portion of the limited access route is through densely populated built-up areas. The Container Terminal of Galveston, Inc. is served by four major railroads: the Burlington Northern, Santa Fe, Southern Pacific, and Union Pacific Lines. Galveston Railway, Inc., provides terminal connections and performs switching of all rail traffic. An intermodal container transfer terminal is located within the container terminal and trackage extends to within 30.5 m (100 ft) of ship berths (Jane's, 1992; AAPA, 1993; Schultz, 1993).

Other Pertinent Information: The Port of Galveston has its own security force that provides 24-hour surveillance of its terminals. Container Terminal of Galveston, Inc. is fenced and has controlled access. An area is provided for segregation and temporary storage of hazardous cargoes.

The Port of Galveston's Director of Operations was unaware of any regulations prohibiting the importation of spent nuclear fuel (Schultz, 1993). The port occasionally handles hazardous materials, including Class A explosives (Schultz, 1993). NRC records indicate the port has not handled foreign research reactor spent nuclear fuel since at least 1979 (NRC, 1993).

The container terminal operator is responsible for handling hazardous materials emergencies at the Container Terminal of Galveston, Inc. facility. The Port of Galveston relies on the Galveston Fire Department's hazardous materials team and/or highly trained hazardous materials personnel at refineries located some 16-24 km (10-15 mi) away. The West Gulf Employers Association holds training courses for longshoremen which Port of Galveston terminal personnel also attend (Schultz, 1993).

Galveston is a major resort and tourist center for the Southwest United States. There is a 2.95 ha (7.3 acres) waterfront tourist attraction at "Pier 21" close to the historic district. A hospital is located across the street from the general cargo berths (Schultz, 1993). A public park on Pelican Island, reached by causeway, is located across the Intracoastal Waterway from the port. A cruise ship terminal is located at Pier 25 in the heart of the port complex and there is a tanker terminal on Pelican Island across from the port at its southern end. The greatest source of potential conflict is the heavy tanker traffic utilizing the Galveston entrance channel en route to Texas City and the Port of Houston petroleum/petrochemical terminals. Houston is the third most active port in the United States in terms of tonnage handled

(IPA, 1993). The U.S. Coast Guard accident data for the period 1991-1993 indicate 52 reported accidents in the Galveston Bay area (USCG, 1994b). This includes ship traffic bound for the Houston area and also includes barge accident data.

Other than general heightened environmental awareness, there are no known sensitive environmental areas in the Port of Galveston area (Schultz, 1993). The port is subject to hurricane and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Galveston, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census population within 16 km (10 mi) of the port terminals was 73,322. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 403,000; Oak Ridge Reservation, 337,000; Idaho National Engineering Laboratory, 526,000; Hanford Site, 575,000; and Nevada Test Site, 595,000. Populations along rail routes to these sites are slightly larger for Savannah River Site and Oak Ridge Reservation, but are slightly less for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,600 km (1,000 mi); Oak Ridge Reservation, 1,550 km (963 mi); Idaho National Engineering Laboratory, 3,070 km (1,911 mi); Hanford Site, 3,740 km (2,327 mi); and Nevada Test Site, 3,000 km (1,862 mi). Distances along rail routes are slightly longer.

Environmental Conditions

A large number of aquatic and terrestrial species frequent the Galveston Bay area. A variety of birds migrate, winter, and breed along the Texas Coast including shorebirds, songbirds, waterfowl, and raptors (Breslin, 1993; FWS, 1992). These endangered/threatened bird species include the brown pelican, peregrine falcon, bald eagle, Attwater's greater prairie-chicken, piping plover, and the eskimo curlew (State-threatened only). Endangered/threatened marine mammals and sea turtles also are found in the coastal bay systems and the Gulf of Mexico. Galveston Bay is within the range of the green, hawksbill, Kemp's ridley, leatherback, and loggerhead sea turtles. While no protected species are known to be located within the Port of Galveston, significant populations of the endangered brown pelican and the piping plover exist nearby (Werner, 1994). The U.S. Fish and Wildlife Service reported that as many as 600 brown pelicans have been sighted loafing on the north end of Little Pelican Island, which is approximately 5.6 km (3.5 mi) northwest of the port. In addition, approximately 125 pairs nested and produced 90 young for the year at this site in 1994, the first time that brown pelicans had successfully nested in Galveston Bay in over 40 years. Wintering populations of the threatened piping plover use the northeastern end of Galveston Island and the southeastern end of Bolivar Peninsula. Of the 3,187 birds observed during the first Gulf Coast count of wintering piping plovers, 1,904 were observed on the Texas coastline (Werner, 1994).

A great amount of commercial and recreational fishing occurs in Galveston Bay and the Gulf of Mexico. Shellfish are the most important commercial species, particularly shrimp, followed by eastern oysters and blue crabs (TPWD, 1989a). The most valuable finfish landed from the Galveston Bay system are black drum, and mullet. In 1988, a total of 5,077,170 kg (11,169,773 lbs) of shellfish valued at $13,489,146 was landed from the Galveston Bay System; a total of 224,536 kg (493,980 lbs) of finfish valued at $226,140 was also landed. The major recreational species of fish that were caught in the Galveston Bay system in 1987-1988 were: Atlantic croaker, sand seatrout, spotted seatrout, southern flounder, black drum, and red drum (TPWD, 1989b).

While the port area is highly developed, a wide variety of marine, estuarine, and lacustrine wetlands exist along Galveston Bay, including a large portion of Pelican Island, directly west of the port. Wetlands also occupy the majority of the far northern end of Galveston Island (FWS, n.d.a.).

Climatic Conditions

The City of Galveston is bounded on the southeast by the Gulf of Mexico and on the northwest by Galveston Bay. Thus, the climate of the Galveston area is predominantly marine, with periods of modified continental influence during the colder winter months when cold fronts from the northwest sometimes reach the Texas coast. Because of its coastal location, sub-freezing temperatures are rare, and higher than normal humidities prevail throughout the year. Summer rainfall is highly variable across the island due to thunderstorms and the local sea breeze circulation. Winter precipitation comes mainly from frontal activity and onshore flow, which produces slow, steady rains under a low stratus cloud deck. The island has been subject at infrequent intervals to major tropical storm systems with hurricane-force winds (NOAA, 1993c).

D.2.1.3 Hampton Roads, VA (Includes the Combined Terminals at Newport News, VA; Norfolk, VA; and Portsmouth, VA)

Hampton Roads is one of the world's foremost bulk cargo harbors. It is a multi-terminal port with privately and publicly owned marine cargo handling facilities located at the southwest corner of the Chesapeake Bay at the confluence of the James and the Elizabeth Rivers. The port is about 26 km (16 mi) from the Virginia Capes and the entrance from the Atlantic Ocean. The major terminals located on the Elizabeth and James Rivers are approximately another 10 to 13 km (6 to 8 mi) from the Bay (DOC, 1993c). The port includes the ports and cities of Norfolk, Portsmouth, and Newport News. Adjacent communities include the cities of Chesapeake and Virginia Beach. The maps of the port are shown in Figures D-22 (Newport News), D-23 (Norfolk), and D-24 (Portsmouth).

In 1992, Hampton Roads handled approximately 5.9 million metric tons (6.5 million tons) and 875,000 20-ft equivalent units of containerized cargo, including large amounts of radioactive materials (primarily uranium dioxide). The port ranks closely with the port of Charleston as the second or third most active container port for the East and Gulf Coasts (DOE, 1994d). The port is serviced by more than 75 ship lines that serve the port on a regular basis and provide approximately 4,000 sailings a year to many countries of the world, including Scandinavia, Europe, the Mediterranean, Near East, Mideast, Far East, Africa, Japan, and South America. A partial listing of lines include Alianca, American-Africa-Europe, American Transport, Argentine Line-ELMA, ACL, Atlantic Express, Bank, Ceylon Shipping, CGM, Chilean Line, Cho Yang Shipping, COSCO, DB Turkish Cargo Lines, Deppe, DSR Senator, Eimskip, Evergreen, Farrell, Hapag-Lloyd, Hoegh, Italian Line, Ivarian Lines, Jugolinja, K Line, Lloyd Basilero, Lykes Lines, Maersk, Mediterranean Shipping, Mitsui OSK, NSCSA, Nedlloyd, Neptune Orient, Netumar Lines, NYK, OOCL, Ocean Star Container Line, P & O, PT Djakarta Lloyd, Safbank Lines, Sea-Land Service, Shipping Corp. of India, Spanish Line, Tokai Shipping, Toko Kaiu Kaisha Ltd., Torm West Africa, United Arab Shipping, Venezuelan Line Walleniuis, Waterman, Wilhelmsen, Yang Ming Line, and Zim (Jane's, 1992; AAPA, 1993).

Because Hampton Roads and its approaches from the Virginia Capes handle a large amount of shipping, traffic separation schemes have been established for the control of maritime traffic. The controlling depth in the Deep Water Route from the Virginia Capes is 15.2 m (50 ft), except for one 14.3 m (47 ft) location. Projected depth for the Hampton Roads channel varies from 15.2 m to 16.7 m (50 to 55 ft). Depth alongside terminals at Newport News and Portsmouth is about 11.6 m (38 ft), while at the Norfolk terminal it is 12.5 m (41 ft) (DOC, 1993c; Jane's, 1992).

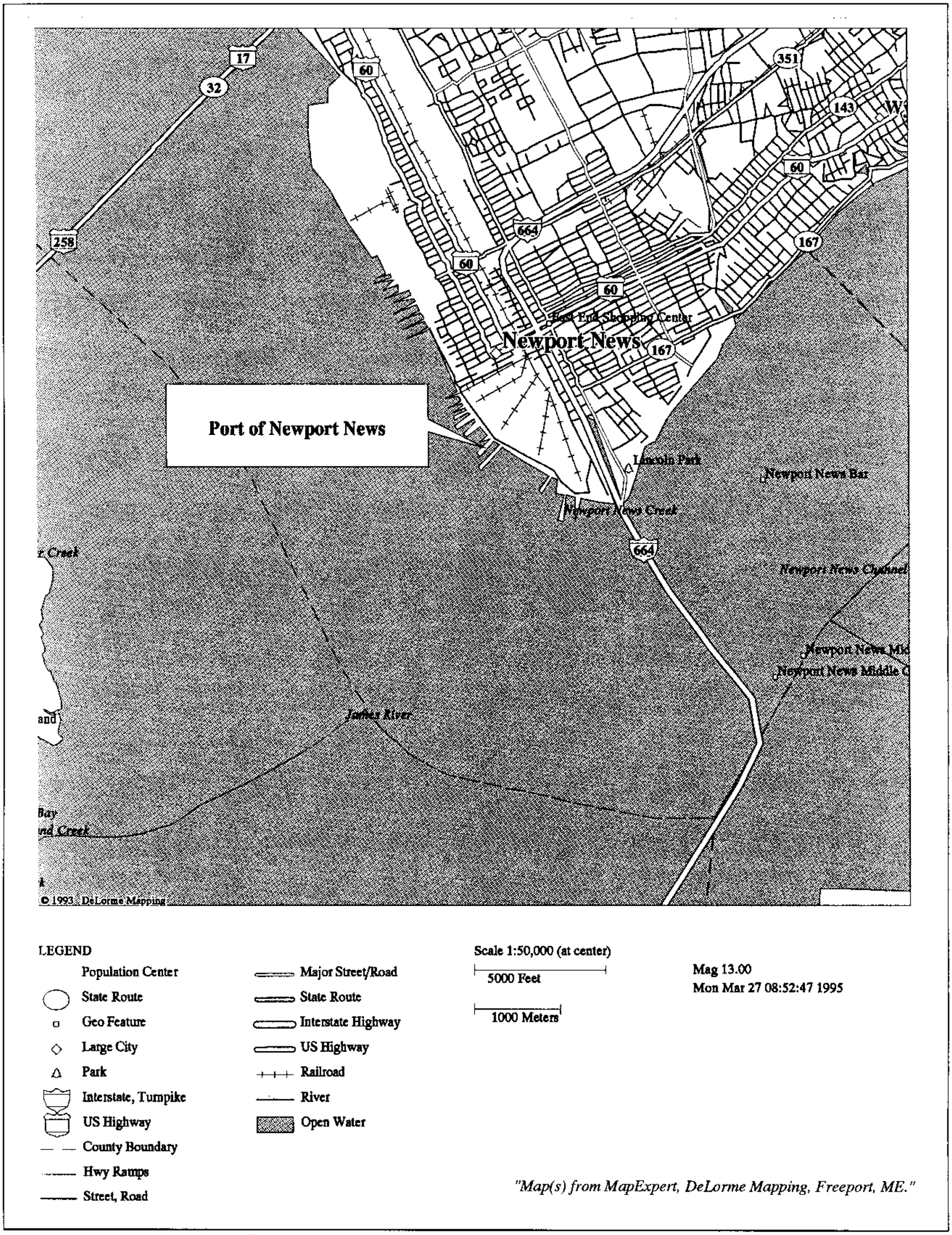


Figure D-22 Map of the Port of Newport News, VA

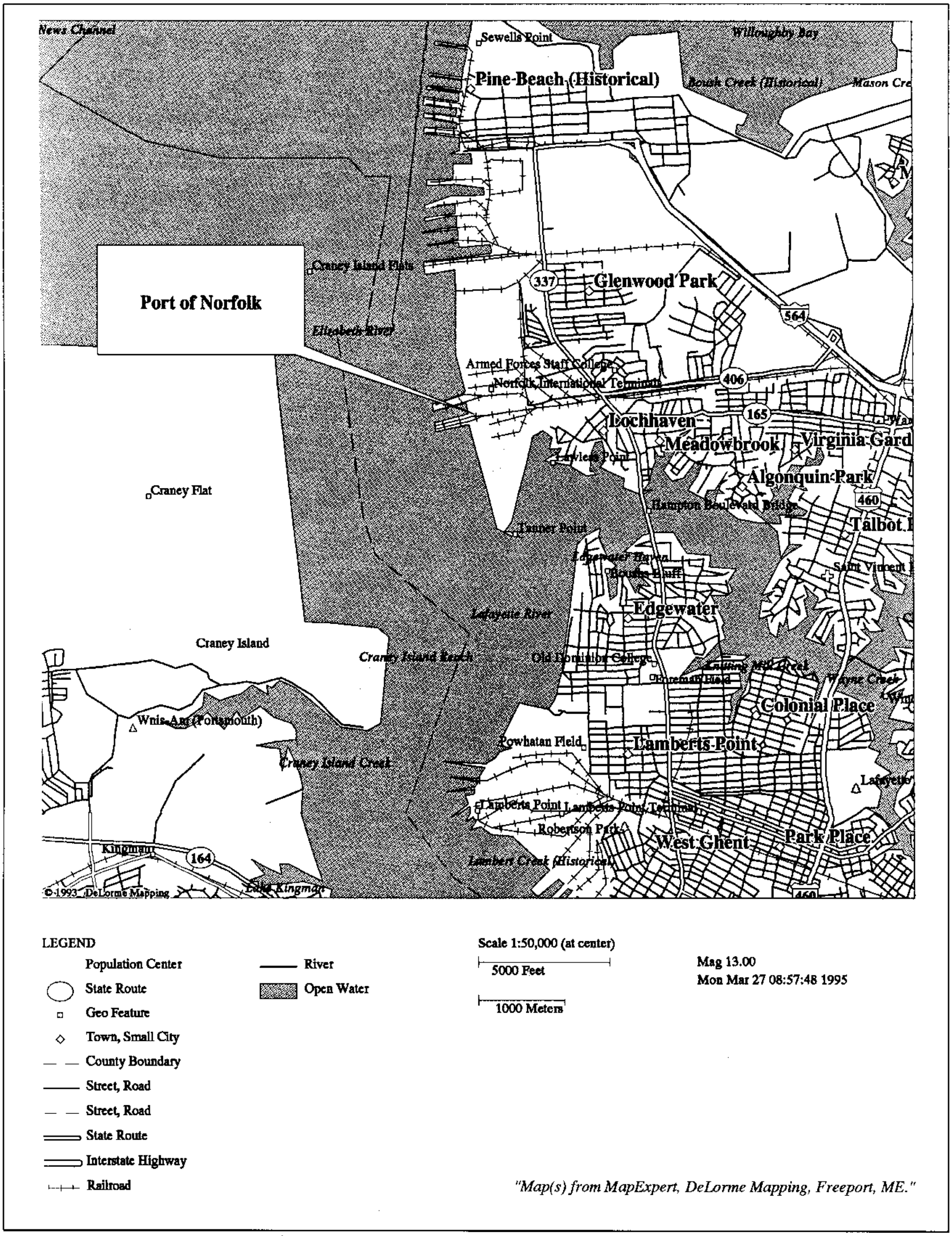


Figure D-23 Map of the Port of Norfolk, VA

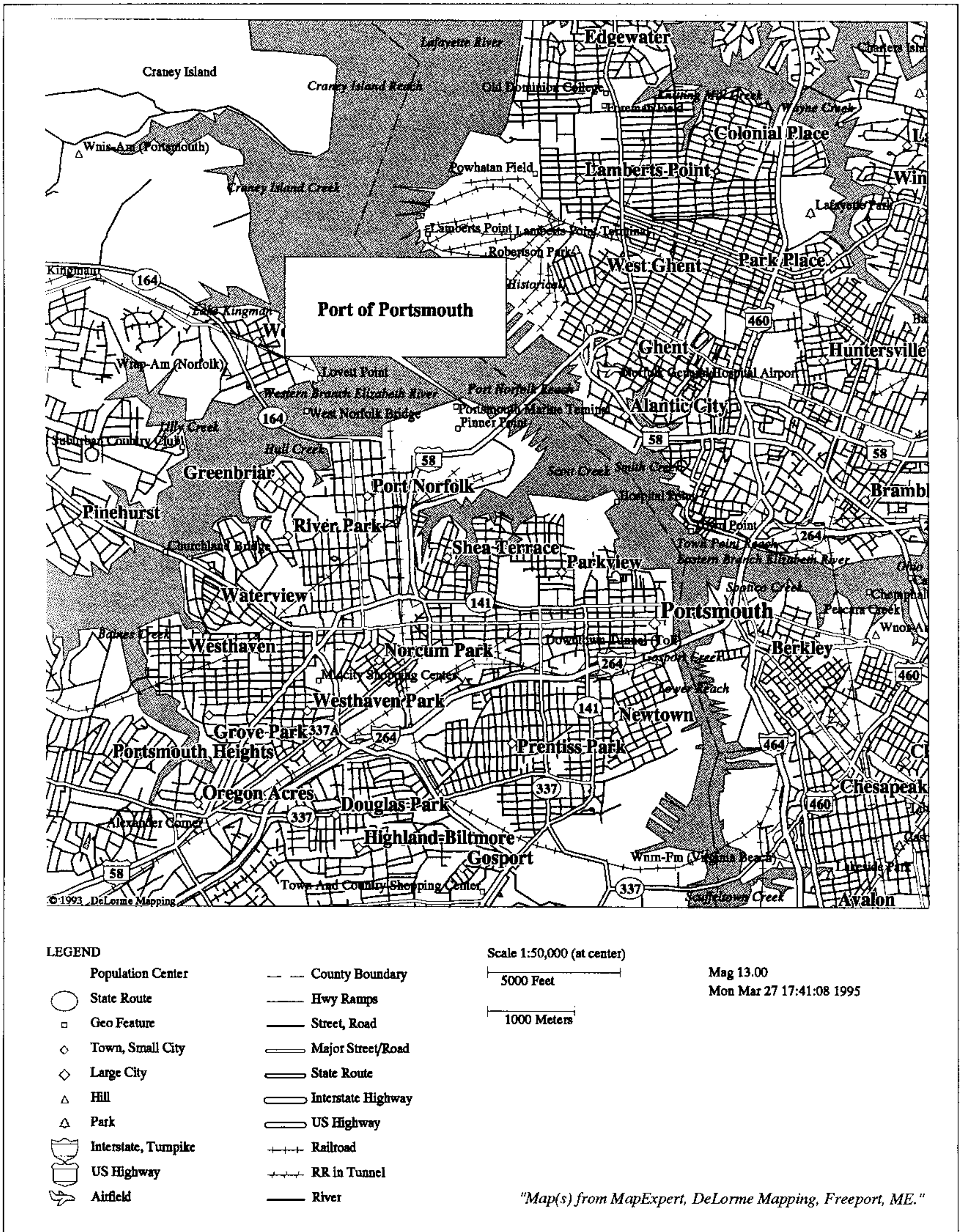


Figure D-24 Map of the Port of Portsmouth, VA

Changing weather can also be a concern as noted in the U.S. Coast Pilot: "Weather deterioration in the lower bay is often sudden and violent and constitutes an extreme hazard to vessels operating or anchoring in this area. The proximity of the bridge-tunnel complex to main shipping channels adds to the danger. Currents in excess of 1.5 meters-per-sec (3 knots) can be expected in this area" (DOC, 1993c).

The presence of three major vehicle tunnels (Chesapeake Bay tunnels, and Hampton Roads Tunnel with associated bridges) under the shipping channels are also sources of risk from ship collisions, especially in fog or during bad weather. Overall, however, the transit is direct and well-managed (DOC, 1993c).

The terminals of primary interest are owned by the Virginia Port Authority, that is a state agency reporting to the Secretary of Economic Development. The Virginia Port Authority's three large, general cargo terminals within the Greater Hampton Roads harbor area include Norfolk International Terminals which is a large container port that includes Sewell's Point Docks (a breakbulk facility), Portsmouth Marine Terminal, and Newport News Marine Terminals. These Terminals are operated by the Virginia International Terminals (the operating arm of the Virginia Port Authority). Lambert's Point Docks, a large breakbulk terminal owned by Norfolk Southern Railroad is also located on the Norfolk waterfront, but lacks container cranes. All three terminals are located in commercial port districts of their respective cities, somewhat separated from other community activities, in areas dedicated primarily to port industrial usage. The three Virginia Port Authority terminals are discussed below in subsections by terminal (Jane's, 1992; AAPA, 1993; FHI, 1994b; DOE, 1994d; VPA, 1994).

Other Pertinent Information: There are no regulatory restrictions prohibiting the receipt and handling of spent nuclear fuel in the port. Compliance with hazardous materials regulations (49 CFR) is the primary requirement. The Portsmouth Marine Terminal has had extensive experience in the receipt and handling of spent nuclear fuel shipments in the recent past, and Norfolk International Terminal and Newport News Marine Terminal also have had some experience (SNL, 1994; NRC, 1993). There appears to be little or no conflict with other hazardous cargoes, including petroleum products, naval weapons depots, etc., in the immediate vicinity of the three Virginia Port Authority terminals. The Virginia Port Authority depends on the Hampton Roads Emergency Team for response to hazardous materials accidents within its terminals. Hampton Roads Emergency Team consists of the fire departments of Norfolk, Portsmouth, and Virginia Beach, in liaison with the U.S. Coast Guard. Chief White of the Portsmouth Fire Department is in charge of the team, which also has ties to the State Emergency Team. All of the Virginia Port Authority terminal operating personnel and longshoremen are currently trained in hazardous materials awareness. Security for the port is provided by perimeter fences and the Virginia Port Authority's Port Police, which maintain 24-hour patrol and surveillance at all three terminals. The state of Virginia's Safety Manual sets forth the rules and policies for operations, including, among other things, hazardous cargoes, container control, emergency procedures and general safety, and provides the policy for receipt and handling of radioactive materials, including emergency response, personnel protection, facility protection, environmental protection and cargo protection (Edwards and Drews, 1993).

All three terminals are located in a large urban area in which congestion is to be expected. Of the three terminals, Portsmouth Marine Terminal is located closest to residential and downtown areas; however, Portsmouth is a relatively small city in both area and population, and it is only a short distance from the terminal to more sparsely populated rural areas. Conversely, truck shipments from Norfolk International Terminals, the terminals closest to the sea, must travel about 38 km (24 mi) of heavily trafficked Interstate through built-up sections of Norfolk, Virginia Beach, and Chesapeake before reaching Bowers Hill (a rural area). The comparable distance from Portsmouth is about 6 km (4 mi).

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Hampton Roads, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (90 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Environmental Conditions

The lower Chesapeake Bay - Hampton Roads area is located on the coastal plain of southeastern Virginia. This area is rather flat and is dissected by numerous bays, rivers, creeks, and wetlands including saltwater marshes, bogs, and swamps (DOE, 1995). However, the areas in the vicinity of the Ports of Newport News, Norfolk, and Portsmouth are highly developed and the waterfronts consist largely of piers and bulkheads associated with the various shipyards, shipping terminals, warehouses and railroad yards that comprise this heavily utilized harbor area. The U.S. Fish and Wildlife Service reports that there are no Federally-listed or proposed listing for endangered or threatened species within a one-mile radius of the Portsmouth or Newport News Terminals (Mayne, 1994). However, rare, threatened, and endangered plant and animal species may be found around the cities of Newport News, Norfolk, and Portsmouth (O'Connell, 1994).

The waters of the Chesapeake Bay and Hampton Roads can be classified as a high-salinity estuarine habitat (generally 16.5-30 parts per thousand), while the Elizabeth and James Rivers, in the vicinity of the ports addressed in this section, are classified as mid-salinity estuarine habitat (generally 5-16.5 ppt) (FWS, 1980d). A number of aquatic species can generally be found in mid-salinity estuarine habitat in this area. Of particular note is the Atlantic sturgeon, a state-endangered species that migrates through these areas. In addition, the eastern oyster is common in the Elizabeth River, in the vicinity of the Norfolk International Terminal and the Portsmouth Marine Terminal. Blue crabs, eastern oysters, and hard clams are also found in the vicinity of the Newport News Marine Terminal (FWS, 1980d). However, Hampton Roads, the Elizabeth River, and portions of the James River (including that portion along the port of Newport News) have been designated as the "Condemned Shellfish Area No. 7" by the Virginia State Department of Health (West, 1994). Shellfishing is either restricted (special permits) or prohibited in this area. The State of Virginia also reports that a fishing health advisory is in effect for the James River and its tributaries due to Kepone contamination. In addition, the Southern Branch of the Elizabeth River, which is upstream of the Norfolk International and Portsmouth Marine Terminals, only partially supports water column standards for dissolved oxygen and sediment standards for lead. The waters in the Hampton Roads area are considered to be "swimmable" by the State of Virginia (West, 1994). The Tidewater area is part of the Mid-Atlantic flyway, and the numerous waterways and wetlands in this area are utilized by many migratory birds that pass through or winter in this region. There is generally a lack of suitable habitat or forage areas in the immediate vicinity of these ports. However, the Ragged Island Wildlife Management Area, located across the James River from the Port of Newport News, is used as a migratory area for waterfowl. Nesting areas for the great blue heron and the yellow-crowned night heron, both State-protected species, are reportedly located on the Lafayette River, approximately 3.2 km (2 mi) upstream of the Norfolk International Terminal (FWS, 1980d). More recently, the U.S. Fish and Wildlife Service indicated that a yellow-crowned night heron rookery consisting of eight nests was documented in the vicinity of the Norfolk Terminal (Mayne, 1994).

There are no known areas of special environmental concern other than the growing interest in preservation of the Chesapeake Bay and its tributary rivers. While the Dismal Swamp National Wildlife Refuge is located about 16 km (10 mi) from the two terminals on the Elizabeth River, the water drainage from the swamps is toward the port, and would not normally carry waterborne radioactivity into the swamp. Further, the swamp is far enough from the terminals so that any radiological impacts from airborne releases (e.g., fires) would be expected to be negligible. In port, any potential negative impacts of low-probability, severe accidents on wildlife populations would be limited to the immediate area around the terminals.

Climatic Conditions

The Port of Hampton Roads, VA is located at the confluence of the James River and the Chesapeake Bay, approximately 29 km (18 mi) west of the Atlantic Ocean. The average elevation of this region is approximately 4 m (13 ft) above sea level.

The geographic location of this region is especially favorable, tending to be located south of the predominant winter extratropical cyclone tracks which originate at higher latitudes and north of the usual tropical cyclone (e.g., tropical storms and hurricanes) paths. In general, the winters are mild with slightly warmer temperatures during the spring and fall seasons. The summer season is warm and long, but is characterized by frequent cool periods, generated by cool northeasterly winds off of the North Atlantic. Extreme cold waves are infrequent, and temperatures below 18°C (0°F) are almost nonexistent. In general, winters pass without measurable snowfall, and most snowfall melts within 24 hrs. The average first sub-freezing day in the fall is November 17 and the last occurrence in the spring is March 23. The predominant wind directions since 1984 are from the south-southwest (about 30 percent) and north-northeast (about 25 percent) and vary seasonally (NOAA, 1992a).

D.2.1.3.1 Newport News Marine Terminal

This terminal is located on the north shore of the Port of Hampton Roads on the James River. It is a combination container, roll-on/roll-off, and breakbulk terminal. The facility has two piers (B and C), a total area of 56.9 ha (141 acres), five berths [two container berths, (each 284 m (930 ft) long), three breakbulk berths (totaling 667 m or 2,190 ft)], and four container cranes [(two 40.6 metric ton (45 ton) and two 30.5 metric ton (33.6 ton)]. Pier B is 189 m (620 ft) long and 168 m (550 ft) wide with three ship berths. Pier C is 285 m (985 ft) long and 165 m (540 ft) wide with equal dimensions of both the north and south sides. There is covered storage on both piers (36,620 m^2, or 394,200 ft^2) and the container handling terminal has storage for 790 stacked containers and 1,210 containers on chassis. The Virginia Port Authority is improving this terminal with a new 9,300 m^2 (100,000 ft^2) warehouse (Jane's, 1992; AAPA, 1993; FHI, 1994b).

Newport News Marine Terminal has immediate access to Interstate-664 outside the terminal. I-664 connects with I-64 Northbound, bypassing the Hampton Roads Tunnel, en route to the Richmond bypass, I-295 South, which connects with I-95 and I-85 Southbound. The terminal is served shipside via CSX Railroad, with direct rail service (Jane's, 1992, FHI, 1994b; AAPA, 1993).

The 1990 population within 16 km (10 mi) of the port terminals was 430,757. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 181,000; Oak Ridge Reservation, 209,000; Idaho National Engineering Laboratory, 628,000; Hanford Site, 677,000; and Nevada Test Site, 691,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 840 km (519 mi); Oak Ridge Reservation, 890 km (553 mi); Idaho National Engineering Laboratory, 4,010 km (2,492 mi); Hanford Site, 4,680 km (2,908 mi); and Nevada Test Site, 4,172 km (2,595 mi). Distances along rail routes are slightly longer.

D.2.1.3.2 Norfolk International Terminals

This is the Virginia Port Authority's largest container handling facility, located on the south side of the Port in Norfolk, adjacent to the Navy Base on the Elizabeth River Channel. The Norfolk International Terminals have a wharf area of 328 ha (811 acres), 4 container berths, 7 container cranes, room for stacking 23,930 20-ft equivalent units four high, chassis stackers for 702 chassis, a roll-on/roll-off berth and covered pier storage of 83,640 m^2 (900,000 ft^2) plus associated container terminal handling equipment. Sewell's Point Terminal, located at the north end (seaward) of the Norfolk International Terminals' container berths consists of 12.14 ha (30 acres) of land area, two piers, and covered storage for breakbulk cargoes. The Norfolk International Terminals have four container berths, 1,289 m (4,230 ft) in length, six 40.6 metric ton (45 ton) container cranes, and one 30.5 metric ton (33.6 ton) container crane.

The terminal is located approximately 2.9 km (1.8 mi) from Interstate 64 via International Terminal Boulevard (a multi-lane industrial roadway bordering the Norfolk Naval Base). It is assumed that travel on I-64 would be southbound only because of the Hampton Roads Tunnel on I-64 North. Southbound routing requires crossing several bridges over the Eastern and Southern Branches of the Elizabeth River and dealing with frequent traffic congestion on the heavily traveled Interstate. The Norfolk International Terminals is served directly (shipside) by the Norfolk Southern Railroad, and indirectly via the Norfolk and Portsmouth Belt Line Railroad, with CSX and Eastern Shore Railroads (AAPA, 1993; FHI, 1994b).

The 1990 population within 16 km (10 mi) of the port terminals was 681,864. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 131,000; Oak Ridge Reservation, 174,000; Idaho National Engineering Laboratory, 631,000; Hanford Site, 694,000; and Nevada Test Site, 694,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 800 km (498 mi); Oak Ridge Reservation, 880 km (550 mi); Idaho National Engineering Laboratory, 4,070 km (2,530 mi); Hanford Site, 4,740 km (2,949 mi); and Nevada Test Site, 4,240 km (2,633 mi). Distances along rail routes are slightly longer.

D.2.1.3.3 Portsmouth Marine Terminals

This is the Virginia Port Authority's second largest marine container handling facility, located further upstream at the confluence of the Elizabeth River and its Western Branch in the City of Portsmouth. The terminal has 3 berths that handle container, breakbulk and roll-on/roll-off cargoes, and a total land area of 88.7 ha (219 acres). It has four marginal berths with a total length of 1,080 m (3,540 ft), with 759 m (2,490 ft) of container berths. The terminal has a storage capacity of 1,770 stacked containers and 2,000 containers on chassis. The terminal also has 14,900 m^2 (160,400 ft^2) of warehouse space. The terminal has three 30.5 metric ton (33.6 ton) container cranes, one 40.6 metric ton (45 ton) container crane, and one 48.8 metric ton (54 ton) container crane.

The Portsmouth Marine Terminals are located approximately 4 km (2.5 mi) from the entrance ramp to I-264, a beltway that links up with U.S. Route 58 westbound in the rural Bowers Hill area of Chesapeake en route to I-95 or I-85 south. The assumed route from the Portsmouth Marine Terminals to the Beltway would be via Harbor Drive and Turnpike Road (State Highway 337), which runs through an area of mixed, small businesses and low-density housing for about 1.6 km (1.0 mi). The Portsmouth Marine Terminals are served directly (shipside) by the CSX Railroad with connections to the other rail lines via the Norfolk and Portsmouth Belt Line Railroad (AAPA, 1993; FHI, 1994b).

The 1990 population within 16 km (10 mi) of the port terminals was 665,700. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 135,000; Oak Ridge Reservation, 257,000; Idaho National Engineering Laboratory, 670,000; Hanford Site, 718,000; and Nevada Test Site, 732,000. Populations along rail routes to these sites are about the same for Eastern sites and slightly larger for Western sites. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 810 km (501 mi); Oak Ridge Reservation, 780 km (487 mi); Idaho National Engineering Laboratory, 4,040 km (2,514 mi); Hanford Site, 4,710 km (2,930 mi); and Nevada Test Site, 4,210 km (2,617 mi). Distances along rail routes are slightly longer.

D.2.1.4 Jacksonville, FL

The Port of Jacksonville is located on the Atlantic Coast of Northern Florida, along the St. Johns River. It is a geographically large city (1,967 km^2 or 760 mi^2), ranging from the town of Orange on the east side of the river to Julington Creek on the west side. Most of the marine terminals are on the west side of the river, about 34 km (21 mi) from the ocean entrance. However, the Blount Island container terminal is well-separated from the city, and is only about 11 km (7 mi) from the harbor entrance. A Federal Project maintains a channel depth of 12.2 m (40 ft) to 12.8 m (42 ft) at the entrance to the river. The depth gradually decreases to about 9.1 m (30 ft) at the railroad bridge in Jacksonville. The Blount Island Terminal is located downstream from the railroad bridge in a deeper part of the channel (DOC, 1993d; Jane's, 1992; AAPA, 1993; Southern Shipper, 1993). A map of the port is shown in Figure D-25.

The St. Johns River has a deep, steep-sided channel cut through the rock in some areas. This channel configuration, combined with the increased size and draft of vessels entering the port makes the river difficult to navigate. Tidal currents in the river are strong as far as Jacksonville, approaching 1.5 meters-per-sec (3 knots) in several places (DOC, 1993d).

The Jacksonville Port Authority (Jaxport) operates two deep water container/general cargo terminals: Blount Island, located approximately 11 km (7 mi) from the harbor entrance and Talleyrand Docks and Terminals, located about 34 km (21 mi) from the entrance. Both terminals are equipped with modern entrance cranes, handle breakbulk and other types of cargo, and have transit sheds, warehouses, and open storage areas. Of the two, Blount Island is preferred because of its separation from the high density downtown area and closer proximity to the sea. A new terminal is under consideration adjacent Blount Island at Dames Point (Southern Shipper, 1993).

Both terminals serve a number of major general cargo and container ship lines from around the world including Sea-Land, NYK, Hyundai, and Mitsui OSK, that offer worldwide cargo services, and Columbus and Blue Star Line (Australia service). These lines provide service to many regions of the world, including Europe, the Mideast, South America, and Australia (Southern Shipper, 1993; Jane's, 1992).

Blount Island Terminal: Blount Island is a 356 ha (880 acre) facility with 1,920 m (6,299 ft) of berthing space, of which Berth 12 is the longest [351 m (1,150 ft)]. Blount Island Berths 7-13 have 11.6 m (38 ft) of water alongside at mean low water, and five 40.6 metric ton (45 ton) container cranes. It has 34,000 m^2 (360,000 ft^2) of transit sheds/warehousing and 149 ha (367 acres) of open storage. This terminal is connected to the mainland via a fixed highway bridge that joins State Highway 105 (Necksher Drive) and connects with I-95 and Route 17 about 8 km (5 mi) north of the City of Jacksonville. A new eight lane truck security plaza was dedicated in 1992. Blount Island has pierside service by the CSX Railroad that connects with the Norfolk Southern Railroad (Southern Shipper, 1993).

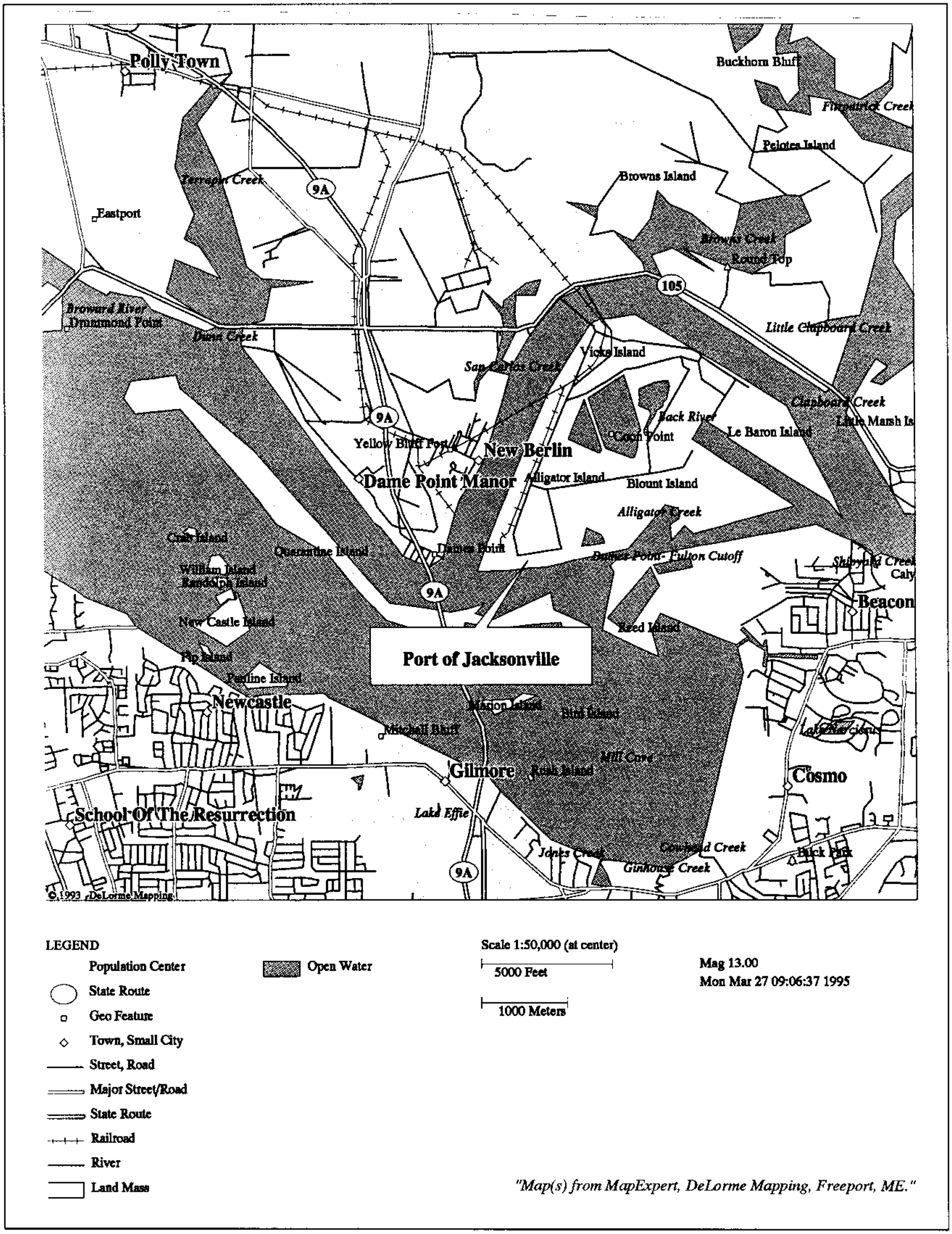


Figure D-25 Map of the Port of Jacksonville, FL

Talleyrand Terminal: Talleyrand Docks is a 70 ha (172 acre) facility with 1,250 m (4,100 ft) of marginal wharf on deep water [11.6 m (38 ft)] at mean low water. It has two 40.6 metric ton (45 ton) container cranes, and two large gantry whirley cranes [50.8 metric ton (56 ton) and 102 metric ton (112 ton)], and 14,900 m^2 (160,000 ft^2) of transit sheds/warehousing with 49 ha (120 acres) of paved open storage (fenced and lighted). Talleyrand Terminal is located in downtown Jacksonville's shopping and commercial zone, about 2.9 km (1.8 mi) downstream of the John R. Matthews Bridge (alternate U.S. Route 90), and less than 1 km (0.6 mi) via city streets to the Expressway (alternate U.S. Route 1) (Southern Shipper, 1993).

Other Pertinent Information: The Port Authority is not aware of any local regulatory restrictions on receipt and handling of spent nuclear fuel (Castiel, 1993). The terminals have no prior experience handling spent nuclear fuel (SNL, 1994; NRC, 1993) or hazardous wastes, but do handle hazardous cargoes such as poisons, corrosives, and Class B explosives. Jaxport is a member of Jacksonville Spillage Control and the City of Jacksonville's Hazardous Materials Team. Terminal operating personnel and longshoremen receive basic instruction in the handling of hazardous cargoes. Around the clock security is provided to both terminals by the Jacksonville Port Authority, with secure, short-term storage available if needed.

There are several tanker terminals and petroleum storage depots downstream and immediately adjacent Talleyrand Docks and Terminals. Blount Island Terminal appears to have no petroleum terminals or other conflicting cargo activities (Castiel, 1993).

While the entire State is environmentally aware, there are no known sensitive wildlife sanctuaries in the immediate area of Jaxport. Blount Island is surrounded by extensive marsh and wetlands. The port is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Jacksonville, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 334,212. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 46,900; Oak Ridge Reservation, 175,000; Idaho National Engineering Laboratory, 576,000; Hanford Site, 643,000; and Nevada Test Site, 639,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 607 km (377 mi); Oak Ridge Reservation, 912 km (567 mi); Idaho National Engineering Laboratory, 4,030 km (2,504 mi); Hanford Site, 4,700 km (2,924 mi); and Nevada Test Site, 4,190 km (2,607 mi). Distances along rail routes are about the same.

Environmental Conditions

The area between the mouth of the St. Johns River and Blount Island is characteristic of typical coastal lowlands found along the southeastern United States. Numerous creeks meander through large expanses of marshes and swamps. With the exception of the U.S. Naval Station Mayport and the village of Mayport, which occupy the first several kilometers along the southern bank of the river, the land bordering the lower portion of the river is largely undeveloped, with the exception of riverfront residences, mainly along the northern bank. In fact, most of the land to the north of the river between Blount Island and the coast is part of the Nassau River - St. Johns River Marshes Aquatic Preserve. The Fort Caroline National Memorial is located southeast of Blount Island on the southern bank of the river. The Little Talbot Island State Park is located approximately 1.6 km (1 mi) north of the channel entrance.

The lower 24.2 km (15 mi) of the St. Johns River has been designated as critical habitat for the manatee, a listed endangered species. The river is also used as a migratory area for the shortnose sturgeon, a listed endangered species (FWS, 1980e). According to the Florida Natural Areas Inventory, the following rare species have been reported within 3.2 km (2 mi) of the Blount Island Terminal: West Indian Manatee (State and Federal Listed Endangered Species), shortnose sturgeon (State and Federal Listed Endangered Species), Atlantic sturgeon (State Listed Species of Special Concern and Federal Listed Threatened Species), sea lamprey, and the opossum pipefish (Murray, 1994). In addition, the U.S. Fish and Wildlife Service reports that the following protected marine species may occur in Duval County: west indian manatee (endangered), shortnose sturgeon (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), hawksbill sea turtle (endangered), and the green sea turtle (threatened). Protected bird species include the bald eagle (endangered), wood stork (endangered), piping plover (threatened), and red-cockaded woodpecker (endangered) (Bentzien, 1994).

A variety of wading birds are also found in the vicinity of the Fort Caroline National Memorial. Several species of birds, including shorebirds, waterfowl, and gannets frequent the area around the jetties at the channel entrance. In particular, the brown pelican (a State Species of Special Concern) is found in this area. A variety of birds inhabit the Little Talbot Island State Park, including the American oystercatcher (a State Species of Special Concern). Loggerhead sea turtles (a listed endangered species) use the beaches along this portion of Florida as a nesting area (FWS, 1980e).

Climatic Conditions

The Port of Jacksonville, FL, is located along the lower 39.4 km (24.5 mi) of the St. Johns River. The terrain in this area is relatively level, providing very little change in relief proceeding inland from the coastal region.

As with the other more northern ports, the climate of this area is also modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the Jacksonville area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months, becoming more southwesterly during spring and summer. Although Jacksonville is along the eastern United States coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years that have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving over land prior to reaching the Jacksonville area. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992e).

D.2.1.5 Military Ocean Terminal, Sunny Point, NC

The Military Ocean Terminal at Sunny Point (MOTSU) is a defense transportation facility used to move military cargo (principally munitions) into and out of the United States. The terminal is located approximately 16 km (10 mi) upstream from the mouth of the Cape Fear River on the Atlantic Coast near Southport, NC. A map of the port is shown in Figure D-26. The port is easily accessed from the ocean, and all commercial vessels bound for Wilmington, NC must pass by MOTSU. It is served by a 12.1 m (40 ft) deep by 152 m (500 ft) wide channel from the ocean (DOE, 1994d).

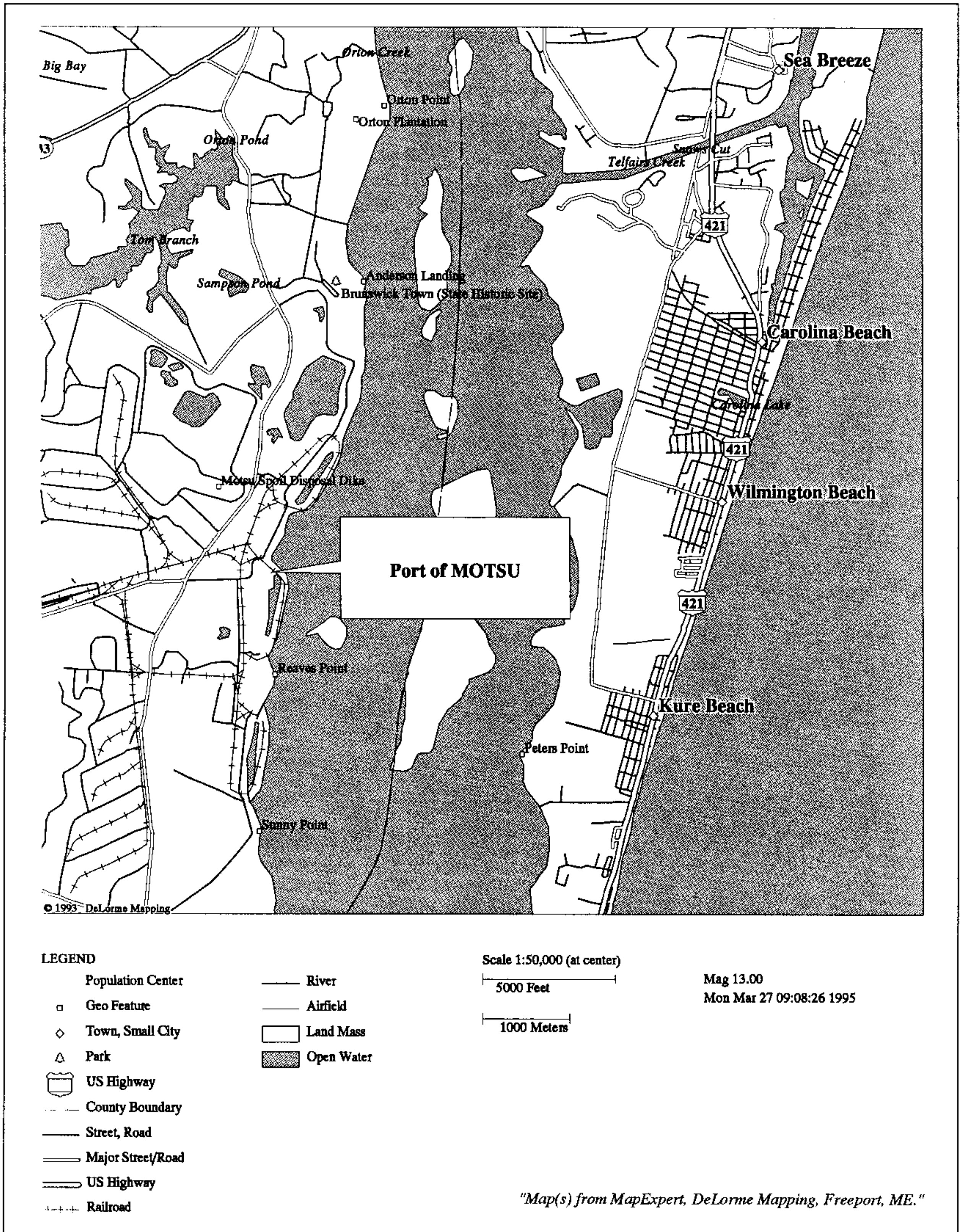


Figure D-26 Map of the Military Ocean Terminal, Sunny Point, NC

Since the majority of cargoes handled at MOTSU are explosive, the terminal is laid out such that an explosion at one wharf will not seriously impact activities at an adjacent wharf. This would permit containerized spent nuclear fuel carried in a commercial vessel (without explosive or hazardous cargoes on-board) to be safely received and transported from the terminal, even though there are conflicting activities within the terminal. Further, after many years of service, MOTSU has never had an explosion accident, so the risks are believed to be small. However, unloading of spent nuclear fuel would be scheduled during periods when explosives were not being unloaded. On average, MOTSU receives about 70 vessels per year, and moves approximately 433,000 metric tons (476,000 tons) of cargo through the port.

While regularly scheduled commercial container or breakbulk vessels do not call at MOTSU, commercial container vessels chartered by defense agencies routinely call at the port. The water depth (channel and alongside the wharves) of 10.3 m (34 ft) mean low water is adequate for most commercial breakbulk, roll-on/roll-off, and container ships. The terminal has three 606 m (2,000 ft) wharves, each with three berths. All wharves have three parallel sets of rail tracks. Berth 1, on the south wharf, has two 45.3 metric ton (50 ton) container cranes capable of off-loading container or container/breakbulk vessels. Berth 3 has been modified with a 30 m (100 ft) wide, reinforced concrete apron that permits breakbulk and roll-on/roll-off operations in addition to containerized cargoes (DOE, 1994d).

MOTSU is serviced by well-maintained roads which are primarily two-lane roads providing connections to multi-lane controlled access highways. In the event that MOTSU was utilized for receipt of foreign research reactor spent nuclear fuel, all transport of spent nuclear fuel over these roadways would be in conformance with State regulations for normal truck traffic between MOTSU and other locations to avoid overloading roadways and bridges. Truck access is provided by State Route 87 from the northwest and State Route 133 from the north. Route 87 provides access to U.S. 17, which runs southwest or northeast. The distance from the terminal gate to Route 133 is about 3.7 km (2.2 mi). Route 133 runs directly to U.S. 17 just outside Wilmington, NC. From Wilmington, U.S. 74 runs west 120 km (75 mi) to Interstate 95, the nearest major north-south highway (DOE, 1994d). A dedicated 157 km (97.4 mi) U.S. Army rail line connects the CSX network directly to the terminal.

Other Pertinent Information: At the present time, there are no regulatory restrictions on receipt, handling, and transhipment of foreign research reactor spent nuclear fuel at MOTSU. MOTSU is the only port in the contiguous United States which has current experience with foreign research reactor spent nuclear fuel receipt and handling, with two shipments received in October 1994 under the Urgent Relief Environmental Assessment.

Cargo handling at the terminal, including explosives, is performed by members of the International Longshoremen Association. Port security is maintained on land by security guards, and on water by dedicated patrol boats.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For MOTSU, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 7,995. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 34,200; Oak Ridge Reservation, 128,000; Idaho National Engineering Laboratory, 463,000; Hanford Site, 548,000; and Nevada Test Site, 619,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 402 km (250 mi); Oak Ridge Reservation, 798 km (496 mi); Idaho National Engineering Laboratory, 3,873 km (2,407 mi); Hanford Site, 4,615 km (2,868 mi); and Nevada Test Site, 3,953 km (2,457 mi). Distances along rail routes are slightly longer.

Climatic and environmental information for MOTSU is similar to that for the Port of Wilmington, NC, as listed in Section D.2.1.10.

D.2.1.6 Naval Weapons Station (NWS) Concord, Concord, CA

Concord NWS is located on the western edge of Suison Bay, an esturine area carrying the flows of the Sacramento and San Joaquin Rivers to San Pablo Bay through the Straits of Carquinez. By sea, the transit is approximately 55 km (35 mi) northeast of the Golden Gate Bridge. Concord NWS is about 5 km (3 mi) north of the city of Concord, CA. The wharf at Concord NWS is about 8 km (5 mi) north of the city. The site tidal areas surrounding the pier areas (including small off shore islands) comprise about 3,1000 ha (7,648 acres) to provide a large separation from nearby communities, and security for the site. A map of the area is shown in Figure D-27. Concord NWS is aligned under the Pacific Division Division of the Naval Ordnance Center. Concord NWS is aligned under the Pacific Division of the Naval Ordnance Center. The Pacific Division is located at Seal Beach, CA and the Naval Ordnance Center is located at Indian Head, MD (Yocum, 1994b).

The Station currently is a breakbulk facility (primarily munitions, naval ordnance, and other high explosives), with limited container handling capabilities. Most vessels servicing the facility are self-supporting, with on-board cranes for handling cargo. Concord NWS currently has a 100-metric ton (112-ton) floating crane and a truck mounted mobile 82 metric ton (90 ton) crane used to service the Station as needed (several cranes are mounted on a crane ship that ties up alongside cargo vessels and loads or offloads cargoes). The facility also has a roll-on/roll-off berth for stern ramps, and a substantial barging pier. A $57 million modernization program has been approved for completion by 1999, which will add to Pier 3 two new 36 metric ton (40 ton) container cranes and gantry crane rails outside of the existing pier structures. The improvements will permit more efficient handling of containerized cargo. The Station is very similar to the MOTSU facility, with three well-separated wharves with two berths on each [about 360 m to 370 m long (1,180 ft to 1,220 ft) at Pier 3]. Separation is designed to protect adjacent vessels from severe damage (or additional explosions) in the unlikely event of an explosion on one ship. Like MOTSU, each pier has three parallel rail lines. Depth alongside the breakbulk/container wharves is 10.6 m (35 ft) at mean low water, which is adequate for most breakbulk/container vessels. Vessel size is limited by the height limit of 33 m (135 ft) under the bridges over the Strait, and the width of the channel [about 90 m (300 ft) minimum] (Yocum, 1994a and 1994b).

Truck access to Interstate 680 is about 10 km (6 mi) via State Route 4. The site has about 127 km (79 mi) of paved roads and 165 km (103 mi) of rail tracks. Concord NWS is served by the Union Pacific, Southern Pacific, and Santa Fe Railroads (all of which have mainline tracks through the tidal area) and has the equivalent of a small intermodal railyard in the immediate vicinity of the pier where railcars can be brought after loading on the piers. Rail routes include direct movement to the Hawthorne Army Ammunition Depot in Hawthorne, NV, where spent nuclear fuel could be off-loaded to trucks for direct shipment to the Nevada Test Site (Yocum, 1994b).

Other Pertinent Information: Since it is a military facility and an explosives operating area, the entire pier operations waterfront is surrounded by barbed wire fencing with access through military posted gates. The facility has areas for staging (and short-term secure storage) shipments by truck or rail near the pier areas.

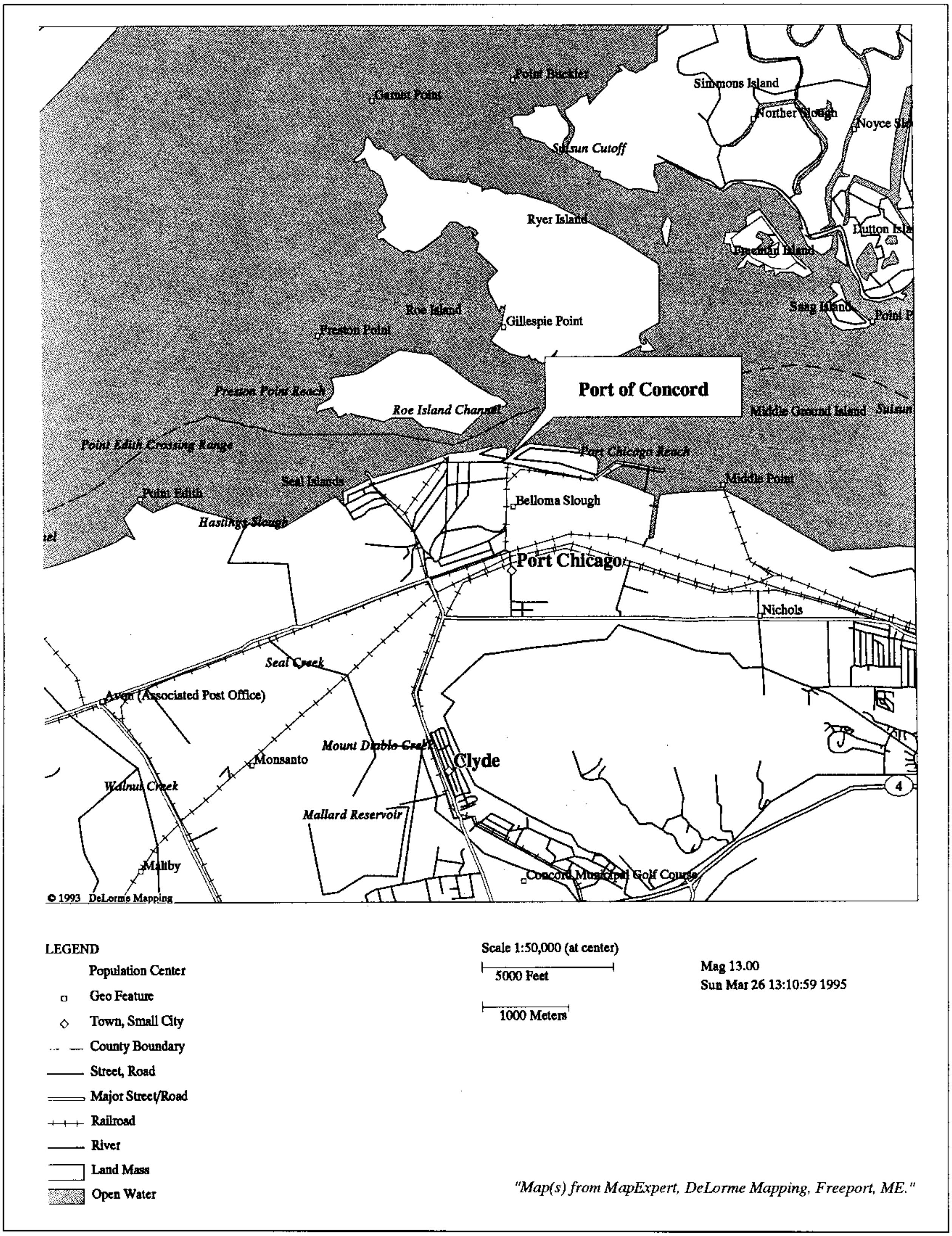


Figure D-27 Map of the Concord Naval Weapons Station, Concord, CA

When explosives are being handled, the explosive safety arc is approximately 3,400 m (11,200 ft) around the pier area. The existing State highway through the site is closed off about 3 km (2 mi) from the piers at the small town of Clyde (population about 485) adjacent to the Station's Administrative areas (Yocum, 1994b). Concord NWS area has its own full-time security force, with a U.S. Coast Guard facility onsite to provide some explosive oversight services during loading and unloading activity. There is a fire station in the immediate vicinity of the pier areas, with an estimated 3-minute response time for first responders.

The primary mission of the port is to support all branches of the military in shipping munitions. No concurrent non-explosives cargo handling, such as foreign research reactor spent nuclear fuel, would be allowed when explosives are being handled. Scheduling of foreign research reactor spent nuclear fuel shipments would have to be done for times when no explosives handling is anticipated. Unscheduled activities or activities with little advance notice involving the military mission would require re-scheduling or re-routing of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel handling would not be the first priority of the port.

Conflicting activities are expected to be avoided by proper scheduling (normally only one ship at a time is in port).

Parts of the tidal area are leased to local cattle growers to keep the grass down for fire protection purposes. The station is a wildlife sanctuary for migratory birds (about 1,200 ha or 3,000 acres) of the tidal area) and hosts native Tule elk, which were formerly on the endangered species list (Yocum, 1994b).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Concord NWS, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located on the edge of a very high seismic zone with an acceleration of 0.45 g.

The 1990 population within 16 km (10 mi) of the port terminals was 381,070. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,040,000; Oak Ridge Reservation, 742,000; Idaho National Engineering Laboratory, 271,000; Hanford Site, 263,000; and Nevada Test Site, 437,000. Populations along rail routes to these sites are slightly smaller for Oak Ridge Reservation, Idaho National Engineering Laboratory and Nevada Test Site, and slightly larger for Savannah River Site, and Hanford Site. These populations are shown in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,476 km (2,784 mi); Oak Ridge Reservation, 4,111 km (2,557 mi); Idaho National Engineering Laboratory, 1,516 km (943 mi); Hanford Site, 1,376 km (856 mi); and Nevada Test Site, 1,145 km (712 mi). Distances along rail routes are about the same for Idaho National Engineering Laboratory, and slightly longer for Savannah River Site, Oak Ridge Reservation, Hanford Site, and Nevada Test Site.

Environmental Conditions

Concord NWS occupies 5,233 ha (12,931 acres) of land adjoining south Suison Bay. Of this total acreage, 2,135 ha (5,276 acres) are inland, while 3,097 ha (7,653 acres) are more tidal in nature. Wetlands comprise approximately 1,215 ha (3,002 acres) of the tidal area (Yocum, 1994b). Wetlands occupy large areas of land bordering all sides of Suison Bay and Grizzly Bay, which is located directly north of Suison Bay. The waters of Suison Bay are characterized as a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). Chinook salmon (endangered), steelhead trout, striped bass, sturgeon, and American shad are typically found in this area (FWS, 1981e; FWS, 1981f).

Portions of the inland area at Concord NWS serve as a sanctuary for Tule elk, a formerly endangered species (Yocum, 1994b). Other terrestrial species found in the area include the river otter, the salt-marsh harvest mouse (a Federally protected species), and the white-tailed kite (FWS, 1981e; FWS, 1981f). Adult concentrations and nesting areas of the California clapper rail (a Federally protected bird species) and the California black rail (a State protected species) are also found in this area. The U.S. Fish and Wildlife Service reports that the following Federally-listed, protected species may occur in Contra Costa County: winter-run chinook salmon (endangered), delta smelt (threatened), bald eagle (endangered), American peregrine falcon (endangered), Aleutian Canada goose (threatened), California brown pelican (endangered), California clapper rail (endangered), California least tern (endangered), and the salt marsh harvest mouse (Medlin, 1994). The Federally and State protected figwort plant family is also found in the vicinity of Concord NWS. In general, the greater San Francisco Bay area annually supports large numbers of shorebirds, wintering waterfowl, raptors, seabirds, and passerlings. In addition, shorebirds, wading birds, waterfowl, seabirds, and songbirds migrate through this coastal area.

Climatic Conditions

Currently, there is no operational National Weather Service station located in Concord, CA. However, the National Weather Service does operate stations at the San Francisco International Airport (37° 37' N; 122° 23' W) and at Stockton, CA (37° 54' N; 121° 15' W). Because of the influence of the California Coast Ranges, which exist between San Francisco and Concord (trending northwest-southeast), the National Weather Service data at Stockton, CA, is considered a better surrogate for the climatological conditions at the Concord Naval Weapons Station.

Concord is located on the westernmost edge of the Great Valley of California, in the eastern foothills of the Coast Ranges. The region is comprised of rich agricultural land, located on the broad delta formed by the confluence of the Sacramento and San Joaquin Rivers. Well to the east of this region are the foothills of the Sierra Nevada Mountains and to the west are the California Coast Ranges. The coast ranges are important in providing an effective barrier from the maritime air masses that greatly influence the San Francisco-Oakland area. However, several gaps in the Coast Ranges do permit the inland migration of the sea breeze circulation, which tends to moderate daytime high temperatures in the summer months. In general, the area is characterized in summer by warm, dry days and relatively cool nights with clear skies and little rainfall. Winter brings relatively milder temperatures, with light precipitation and frequent heavy fog events, which often have long durations in December and January. Ninety percent of the precipitation falls between November and April, with thunderstorms extremely infrequent (4 days per year) and snowfall almost nonexistent (NOAA, 1992j).

D.2.1.7 Portland, OR

The Port of Portland is located about 160 km (97 mi) above the mouth of the Columbia River on the Willamette River tributary. Portland is the principal city of the Columbia River system and one of the major ports on the Pacific Coast. The preferred container terminal (T6) is located approximately 170 km (90 mi) from the entrance of the Columbia River. Federal project depths in the Columbia River are 14.6 m (48 ft) over the bar, and 12 m (40 ft) to Portland (DOC, 1992b). However, a port official indicated the actual channel depth is 13.11 m (43 ft), and the channel width is 183 m (600 ft) from the coast to the port (Magness, 1993).

There are a number of cautions concerning entering and navigating the Columbia and Willamette Rivers. The U.S. Coast Pilot warns that entry into the Columbia River can be dangerous because of sudden and unpredictable changes in the currents often accompanied by breakers. It is reported that "ebb [tide] currents on the [North] side of the bar attain velocities of [3.1 to 4.2 meters-per-sec] 6 to 8 knots . . . In the entrance the currents are variable, and at times reach a velocity of [2.6 meters-per-sec] 5 knots on the ebb; on the flood [tide] they seldom exceed [2.1 meters-per-sec] 4 knots. Since logging is one of the main industries of the region, free floating logs and submerged deadheads or sinkers are also a source of danger. The danger is increased during spring freshets" (DOC, 1992b).

U.S. Coast Guard statistics for 1990 through 1993 indicate that the transit from the Pacific Ocean to the Port of Portland is hazardous, with a reported total of 112 ship collisions and 145 (hard) groundings (USCG, 1994b). It is noted that a large number of oceangoing vessels make the transit on a routine basis without incident. Since some of these accidents were most likely associated with barges, it is believed that the actual rate for oceangoing vessels is probably lower.

The Port of Portland owns and operates Terminal T6, a deep-water dedicated container facility located on Pearcy Island, at the confluence of the Columbia and Willamette Rivers, about 140 km (90 mi) from the ocean entrance to the Columbia River. The port also owns other terminals (including T2, a container/breakbulk facility), all of which lie further upstream of Terminal T6. Terminals are situated in an industrial port district northwest and seaward of downtown Portland (POP, 1994). A map of the port is shown in Figure D-28.

The port is served by several container lines including Australia New Zealand Direct Line, Evergreen Line, Hanjin Shipping Co., LTD., Hawaiian Marine Lines, Hyundai, International Marine Transport Lines, Italian Line, d'Amico Line, Jebson's International, "K" Line, Mitsui OSK, Neptune Orient, NYK Line, Pacific Commerce Line, Safbank Line, and United Yugoslav Line (Jane's, 1992; AAPA, 1993; POP, 1994).

Terminal T6: This terminal has three berths, five container cranes [two 36.3 metric ton (40 ton) and one 50 metric ton (55 ton)], a container freight station, distribution warehouse, and rail/barge service. It has about 869 m (2,851 ft) of marginal wharf, with 12.2 m (40 ft) of water alongside at mean low water. Truck access to Interstate 5 is via North Marine Drive and N. Lombard Street, both of which connect with I-5 about 5.5 km (3.4 mi) from the terminal entrance. North Marine Drive is an industrial use roadway that connects with I-84, the assumed route to Idaho National Engineering Laboratory, north and across the Willamette River from downtown Portland. T6 is served by the Burlington Northern and Union Pacific Railroads, whose tracks reach to within about 0.5 km (1,500 ft) of the container berths; an intermodal, container-on-flat-car rail yard is an integral part of the terminal. T6 has reciprocal switching arrangements with the Southern Pacific Railroad (AAPA, 1993; POP, 1994).

T6 is located north (downstream) of the port's other marine terminals and has no apparent conflict with other hazardous cargoes. It is currently operated by the port, but the port is considering an operations contractor for the future (Hachey et al., 1994).

Terminal T2: This terminal has about 833 m (2,730 ft) of marginal wharf, with 12.2 m (40 ft) of water alongside at mean low water, and four container cranes with capacities ranging from 33 metric tons (36 tons) to over 77 metric tons (85 tons). Truck access to Interstate 84 is via Interstate 5 South to U.S. Route 30 West, connecting with I-84 at Maywood Park, a total distance of about 19 km (12 mi). The terminal is served by the Portland Terminal Railroad and the Burlington Northern, and has direct ship-to-rail transfer capability. T2 also has reciprocal switching arrangements with the Southern Pacific (AAPA, 1993; POP, 1994). T2 is located near several large bulk petroleum terminals that are undoubtedly supplied by tankers. Such traffic was not considered to be a major risk factor for the transportation of spent nuclear fuel to Portland. However, because of the potential conflicting uses, Terminal T6 is the preferred facility.

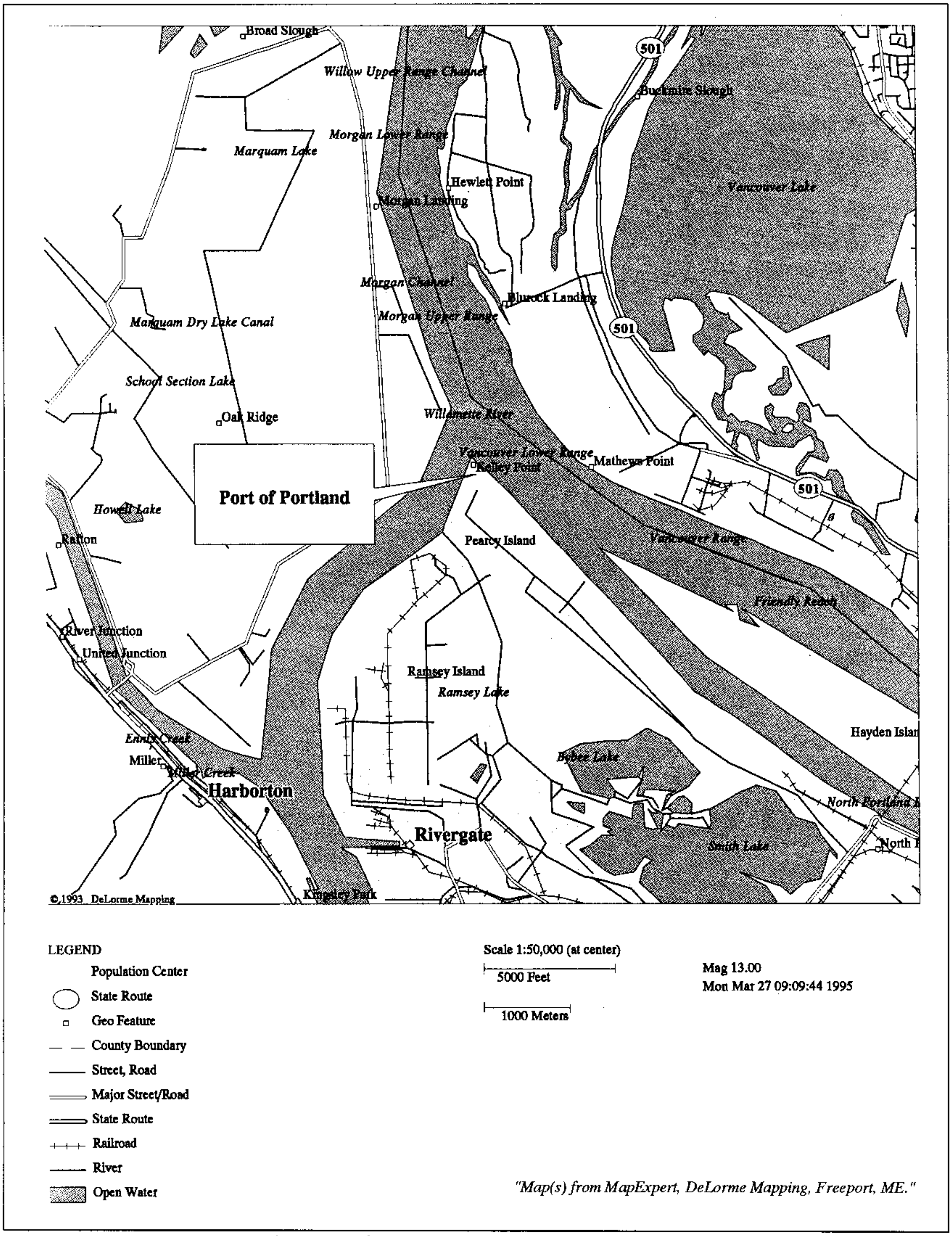


Figure D-28 Map of the Port of Portland, OR

Other Pertinent Information: Security is provided by perimeter fencing and the port's police force, which maintains a 24-hour patrol and surveillance function at both terminals.

There are no restrictive regulations currently affecting the potential receipt and transport of foreign research reactor spent nuclear fuel through the port. The Manager of Marine Market Development indicated that the port has not handled spent nuclear fuel since 1985, and there is opposition to handling nuclear materials by the port's labor unions (Magness, 1993). It is noted that while most of the spent nuclear fuel shipped through Portland had been shipped by the end of 1985, other data sources indicate the port also handled additional spent nuclear fuel in 1989 (NRC, 1993; SNL, 1994). There are no restrictions on Class A or B explosives, and the Coast Guard does not make radiation surveys of radioactive cargoes. Recently, the port could not get shippers to handle naturally radioactive columbium concentrate from British Columbia even if it is not unloaded (Hachey et al., 1994). While this does not preclude foreign research reactor spent nuclear fuel shipments, this indicates there is the potential for delays which could result in failing to "expeditiously transfer" foreign research reactor spent nuclear fuel from the port to a selected storage site.

Portland has a Port Evacuation Plan and a hazardous materials advisory staff (Hachey et al., 1994). The State Health Division with whom the port confers, has a resident nuclear physicist for technical assistance. The port is also a member of the Maritime Fire and Safety Association (an industrial association representing 27 terminal operators) and nine fire departments on the Columbia and Willamette Rivers. The nearest fire station can respond within about six minutes (Hachey et al., 1994). The Association has developed emergency response plans and is implementing a radio communications system covering the entire river system from Astoria to Portland. The City fire department and Coast Guard respond to accidents involving hazardous materials cargoes. Port operating personnel and longshoremen receive general instruction concerning handling of hazardous materials cargoes (Magness, 1993). In addition, the port has contractors ready to respond to hazardous materials accidents when necessary (Hachey et al., 1994). There has not been a severe container accident in at least 10 years, so no port accident statistics were available (Hachey et al., 1994). The port is located several miles downstream from Portland's business and residential districts in an area that appears dedicated to port industrial usage, but as already noted, has excellent connections with highways and rail service.

There are no known areas of special environmental concern in the immediate vicinity of the port, although concern for the environment runs high throughout the Pacific Northwest. A "critical habitat" adjacent to Terminal 6 will have to be mitigated with the planned expansion at T6, but there are no plans to fill wetlands between T6 and populated areas about 1 or 2 km (0.6 or 1.2 mi) away (Hachey et al., 1994).

The port is subject to earthquakes and volcanism. The likelihood of severe natural phenomena, such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Portland, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (90 mph). The port is located in a moderate seismic zone with an acceleration of 0.20 g. There have been two major earthquakes in the Puget Sound area this century; a Modified Mercalli Intensity (MMI) VIII on April 13, 1949, and an MMI VII-VIII on April 29, 1965 (Bolt, 1978). On May 18, 1980, nearby Mount St. Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascade Range are volcanic in origin and prone to eruption (Foster, 1971; Hamilton, 1976; IPA, 1993).

The 1990 census population within 16 km (10 mi) of the Terminal was 356,064. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 686,000; Oak Ridge Reservation, 519,000; Idaho National Engineering Laboratory, 143,000; Hanford Site, 85,700; and Nevada Test Site, 375,000. Populations along rail routes to these sites are slightly smaller for Nevada Test Site and Idaho National Engineering Laboratory, but slightly larger for Savannah River Site, Oak Ridge Reservation, and Hanford Site. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,630 km (2,879 mi); Oak Ridge Reservation, 4,200 km (2,609 mi); Idaho National Engineering Laboratory, 1,190 km (738 mi); Hanford Site, 407 km (253 mi); and Nevada Test Site, 2,040 km (1,270 mi). Distances along rail routes are slightly longer, with the exception of Hanford Site, which is slightly less.

Environmental Conditions

The areas surrounding the Terminal are in river-oriented industrial land use. Wildlife habitat along the Oregon Slough is limited because of the industrial development, although some waterfowl use the area. While the primary uses in the Terminal area are commercial navigation and industry, some recreational fishing and boating occurs in Oregon Slough and the Columbia River (Kurkoski,1994).

The U.S. Fish and Wildlife Service's Ecological Inventory for the Vancouver, Washington-Oregon area indicates that the Columbia River generally includes the following fish species: salmonids, chinook salmon, coho salmon, chum salmon, pink salmon, sockeye salmon, steelhead trout, Dolly Varden, smelts, river lamprey, white sturgeon, American shad, eulachon, and cutthroat trout (FWS, 1981d). South of Portland, the various islands and wetlands along the Columbia River provide habitat for a wide variety of terrestrial organisms. Areas of special interest include the Sauvie Island Game Management Area, which is located approximately 8 km (5 mi) downriver of Terminal 6, and the Ridgefield National Wildlife Refuge, which is approximately 16 km (10 mi) downriver.

The U.S. Army Corps of Engineers reports that raptors such as the red-tail hawk, bald eagle, and peregrine falcon are occasional visitors to this area and that the U.S. Fish and Wildlife Service has indicated that the endangered American peregrine falcon and threatened bald eagle may winter in this area. In addition, the National Marine Fisheries Service has listed the Snake River sockeye salmon as endangered, and two Snake River chinooks stocks as threatened (Kurkoski, 1994). The State of Oregon's Natural Heritage Program reports that there are at least two rare species that occur in the vicinity of Terminal 6 (Gaines, 1994). These species are the painted turtle (a State-Sensitive-Critical species) and the Columbia water-meal.

Climatic Conditions

The city of Portland is situated midway between the northerly oriented low coast range on the west and the higher Cascade range on the east. The Cascade range provides a steep slope for orographic uplift of moisture laden air arriving on westerly winds from over the Pacific Ocean, resulting in moderate rainfall events in the area. The prevailing winds are generally northwesterly during spring and summer, becoming more southeasterly in fall and winter. The Portland area is characterized by a winter rainfall regime, where approximately 88 percent of the annual total falls during October through May. Thus, the winter season is dominated by relatively mild temperatures, cloudy skies and rain accompanied by southeasterly surface winds. Summer produces pleasantly mild temperatures, northwesterly winds and very little precipitation. Fall and spring are traditional seasons with variable characteristics. Fog generally occurs most frequently during fall and early winter. Destructive storms are infrequent in this region of the United States, and surface winds rarely exceed gale force. Thunderstorms occur monthly through the spring and summer, with gentle rains occurring almost daily during the winter months. Based on the 1951-1980 climatology, the first frost occurs on average around November 7, with the last spring frost occurring near April 3 (NOAA, 1992f).

D.2.1.8 Savannah, GA

The Port of Savannah is located on the South Bank of the Savannah River, about 35 km (22 mi) above the entrance from the Atlantic Ocean. Savannah is the third largest city in Georgia, and is the chief port of the State of Georgia. A Federal Project maintains 12.2 m (40 ft) of water through Tybee Roads, then 11.6 m (38 ft) for about 16 mi in the main channel to the turning basin at Kings Island (DOC, 1993d). A map of the port is shown in Figure D-29.

Under normal conditions, currents at the entrance to Savannah are 1.1 to 1.5 metric-sec (2.2 to 3.1 knots) during the ebb tide, and 0.8 to 1.2 metric-sec (1.6 to 2.4 knots) during the flood tide. It has been reported that currents in the river can reach 3.6 to 4.1 metric-sec (7 to 8 knots) in the vicinity of Garden City Terminal just below the Route 17A bridge and at the Colonial Oil Berths, about 4 km (2.5 mi) above the bridge. Access to the port can be complicated due to some relatively narrow sections of the channel combined with high currents (DOC, 1993d).

The Georgia Ports Authority (GPA) operates three large cargo terminals on the South Bank of the Savannah River. Ocean Terminal, located approximately 41 km (25 mi) from the river entrance in the City of Savannah, is a combination breakbulk and container handling facility; Garden City Terminal is about 4.6 km (2.9 mi) further upstream from Ocean Terminal. Containerport, part of the Garden City terminal complex, is a dedicated container handling facility. The depth alongside both container terminals is 11.6 m (38 ft), and dredging to 12.7 m (42 ft) is in progress.

The port is served by more than 50 container and breakbulk ship lines, including several major container carriers, with itineraries to some 100 countries in the world, including many in Europe and the Far East, as well as Japan, and Australia (Jane's, 1992; AAPA, 1993; Southern Shipper, 1993)

Ocean Terminal: This facility has 10 berths, a 61 m (200 ft) apron, extensive Transit sheds and warehouse space, with 34 ha (83 acres) of open storage. It has one 41 metric ton (45 ton) single hoist container crane and four gantry cranes of greater capacity, and 1,825 m (5,990 ft) of marginal wharf; Berth 13, the longest, is 297 m (975 ft) long. The terminal has almost immediate access to U.S. Route 17 (north/south), and connects with I-16 a few city blocks from the terminal. The terminal is served by the Norfolk Southern and CSX railroads (AAPA, 1993; Southern Shipper, 1993; Jane's, 1992). Due to its close proximity to the City, it is not a preferred container terminal.

Containerport: This is the preferred container terminal, due to its better separation from the City and modern facilities. It is located about 40 km (25 mi) from the Atlantic. It has 6 container ship berths, a 61 m (200 ft) apron, and a 457 m (1,500 ft) by 488 m (1,600 ft) turning basin. The terminal has a large Container Freight Station comprising 51,280 m^2 (552,000 ft^2) for stuffing and stripping containers, with areas for segregating hazardous cargoes. The facility has 1,676 m (5,500 ft) of marginal wharf, and nine 40.8 metric ton (45 ton) single hoist container cranes. Truck access to the terminal is via limited access roads and lightly populated areas to the following expressways: about 13 km (8 mi) to I-16 (east/west); about 9.6 km (6 mi) to I-95 north and about 18 km (11 mi) to I-95 south. Containerport has excellent shipside rail service, consisting of a series of rail spurs at right angles to the container berths, providing direct ship-to-rail transfers. Onsite switching is provided by GPA's Savannah State Docks Railroad, connecting with the CSX and Norfolk Southern rail systems (AAPA, 1993; Southern Shipper, 1993; Jane's, 1992; GPA, 1994).

Other Pertinent Information: The port's Director of Public Relations was unaware of any ordinances or regulations prohibiting the receipt and handling of spent nuclear fuel, and the port does handle radioactive materials (Moore, 1993). The official did not know if the port had handled spent nuclear fuel, but data

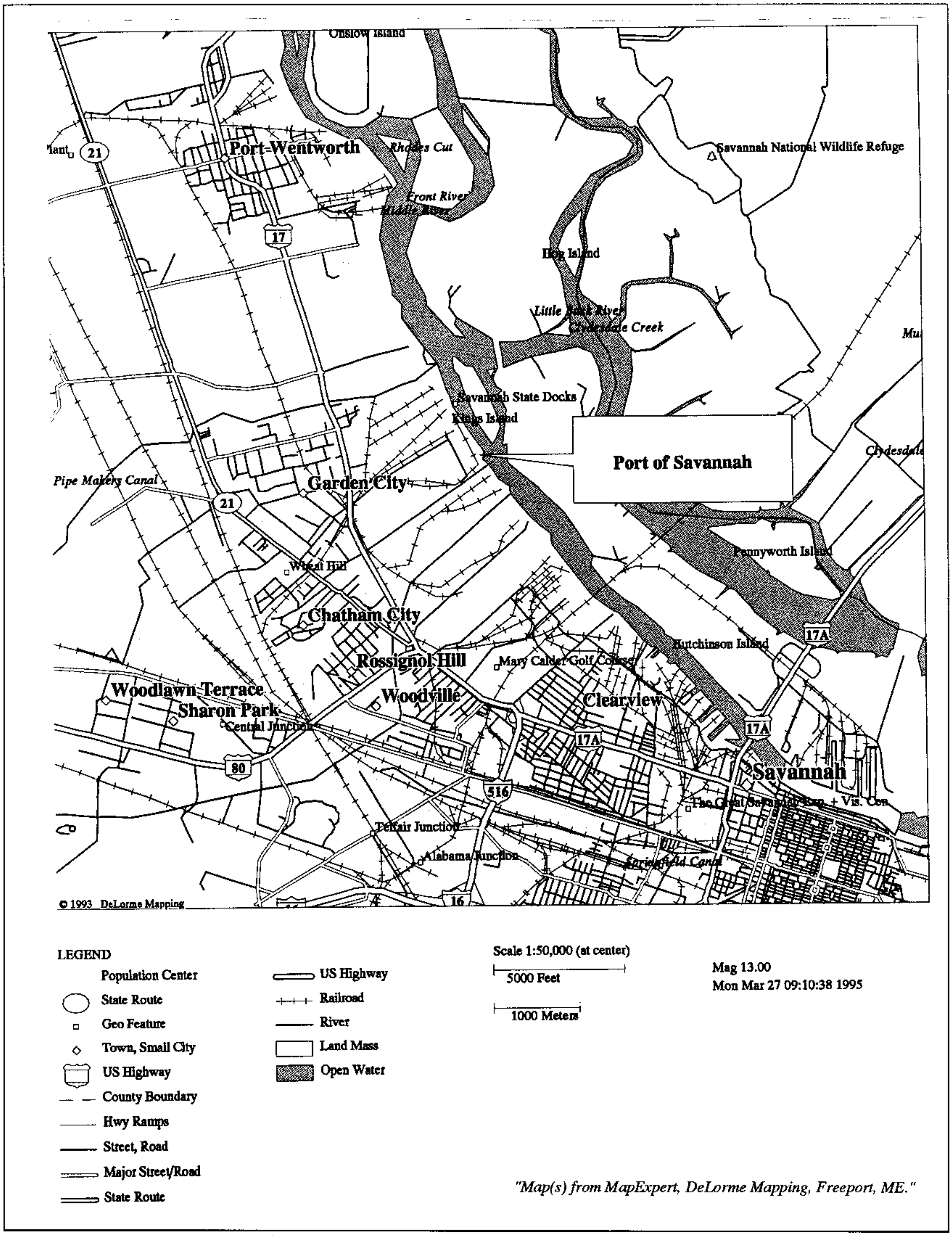


Figure D-29 Map of the Port of Savannah, GA

show it has (SNL, 1994; NRC, 1993). The port has a hazardous materials training staff but no Emergency Response Team. Reportedly, the GPA contracts with outside firms to respond to oil and other hazardous materials accidents.

There are tanker berths and petroleum storage facilities adjacent to Containerport's facilities, and there are several private bulk liquid storage facilities downstream of Containerport (towards the City), including a liquid natural gas terminal a few miles above the Pilot station. The presence of these terminals along a river channel only 152 m (245 ft) wide with swift currents, and the increasing number of container ships with lengths in excess of 250 m (820 ft) heighten the possibility of potentially serious conflicts within the port.

The port is subject to severe hurricanes and tropical storms, and given its proximity to Charleston, SC may have a slightly higher risk of earthquakes than the rest of the State of Georgia. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Savannah, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 155,166. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 37,300 ; Oak Ridge Reservation, 101,000; Idaho National Engineering Laboratory, 553,000; Hanford Site, 602,000; and Nevada Test Site, 616,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 400 km (250 mi); Oak Ridge Reservation, 720 km (449 mi); Idaho National Engineering Laboratory, 3,860 km (2,398 mi); Hanford Site, 4,530 km (2,813 mi); and Nevada Test Site, 4,020 km (2,501 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The lower Savannah River has multiple branches that meander through a variety of coastal lowlands including salt marshes, tidal creeks, freshwater marshes and freshwater impoundments. South Carolina has classified the portion of Savannah Harbor within its boundaries upstream from Fort Pulaski as Class B and the portion oceanward as Class SA. Class B waters are fresh waters suitable for secondary contact recreation, as a drinking water source following conventional treatment, fishing, industrial, and agricultural use. Class SA waters are defined as tidal saltwaters suitable for primary contact recreation, and for all the uses listed in Class B. The State of Georgia has classified the Savannah River from Mi 0 at Fort Pulaski north to Mi 5 at Field's Cut as recreation waters. North of Field's Cut, the waters are classified as Coastal Fishing (U.S. Army, 1991). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Savannah, the Containerport is located in a transitional estuarine habitat where the salinity ranges from low (generally 0.5 to 5 parts per thousand) to mid-salinity (generally 5 to 16.5 parts per thousand) (FWS, 1980c).

A large number of aquatic and terrestrial species are found in and around the Savannah River near Garden City. The U.S. Fish and Wildlife Service indicated that the protected species most likely to be found on or near this general area include: the bald eagle, wood stork, shortnose sturgeon, west indian manatee, and eastern indigo snake. The bald eagle and wood stork occur on the Savannah National Wildlife Refuge. West indian manatees are known to use the lower Savannah River and are seen fairly often in the river and harbor area. The shortnose sturgeon uses the Savannah River as a migratory area. The lower Savannah

River also is reported to be an important spawning area for striped bass (Laumeyer, 1994). In addition, the loggerhead turtle, bald eagle, and the American alligator are found along the lower reaches of the Savannah River (FWS, 1980c).

Both invertebrate and fish species of commercial and recreational value found in the Savannah River. Commercial fishing is primarily for American shad, sturgeon, shrimp, and blue crab. Public shellfishing is allowed in some areas near the mouth of the Savannah River in the vicinity of Fort Pulaski. The Savannah River is host for the migration of several important commercial and game fishes including the American shad, the hickory shad, and the blueback herring. Game species include the spotted seatrout, red drum, croaker, spot, striped bass, flounder, silver perch, white catfish, channel catfish, large mouth bass, sunfish, and crappies. The State of Georgia has closed the striped bass fishery for population recovery purposes. Results of a seasonal creel survey of the Savannah River estuarine fishery, conducted by the Georgia Department of Natural Resources from October 20, 1992 to February 16, 1993 found that the estimated angler harvest for that time period was 10,893 fish. White catfish (28.4 percent), spotted seatrout (27.9 percent), red drum (17.9 percent), and silver perch (10.4 percent) represented approximately 85 percent of the fish harvested from the Savannah River during this time period (Schmitt, 1993).

There are several wildlife refuges and/or game management areas located along the lower portion of the Savannah River. Tybee National Wildlife Refuge is located at the mouth of the Savannah River at the confluence with the Atlantic Ocean. Just north of Tybee National Wildlife refuge is the Turtle Island Game Management Area. The Containerport itself is located across the river from the southern end of the 10,371 ha (25,608 acre) Savannah National Wildlife Refuge. The Savannah National Wildlife Refuge and the Tybee National Wildlife Refuge are managed by the U.S. Fish and Wildlife Service.

Climatic Conditions

The Port of Savannah, GA, is located in Chatham County on the Savannah River. The city of Savannah is surrounded by low, flat terrain that is marshy to the north and east and rises to a few meters (several ft) above sea level to the west and south.

The area has a temperate climate which, again, is greatly influenced by winds coming into the area off of the surrounding ocean. Nominally, 50 percent of the rainfall occurs during thunderstorms with the remainder being equally distributed over the year and generally related to frontal passages. Severe tropical systems affect the Savannah, GA, area roughly once every 10 years and cause heavy, sustained precipitation, high winds, and extreme localized coastal flooding. Rainfall measurements in excess of 51 cm (20 in) have been observed as a result of tropical systems impacting the area. Based on the 1951-1980 climatology, the first freeze occurs on average around November 15th and the last freeze occurs near March 10th (NOAA 1992d).

D.2.1.9 Tacoma, WA

The Port of Tacoma is located in the southeastern corner of Puget Sound on the deep waters of Commencement Bay, about 5 km (3 mi) from the Sound. It is a rapidly expanding major port, second only to Seattle in maritime importance on Puget Sound. Like Seattle, access is gained via the Straits of Juan de Fuca and Puget Sound. The distance from the entrance into Puget Sound is approximately 130 km (80 mi). While the transit is open with deep wide channels, it is a relatively long distance on an inland waterway (DOC, 1992b). The port currently handles about 1,054,000 20-ft equivalent container units, amounting to 6.7 million metric tons (7.4 million tons) of cargo (AAPA, 1994). A map of the port is shown in Figure D-30.

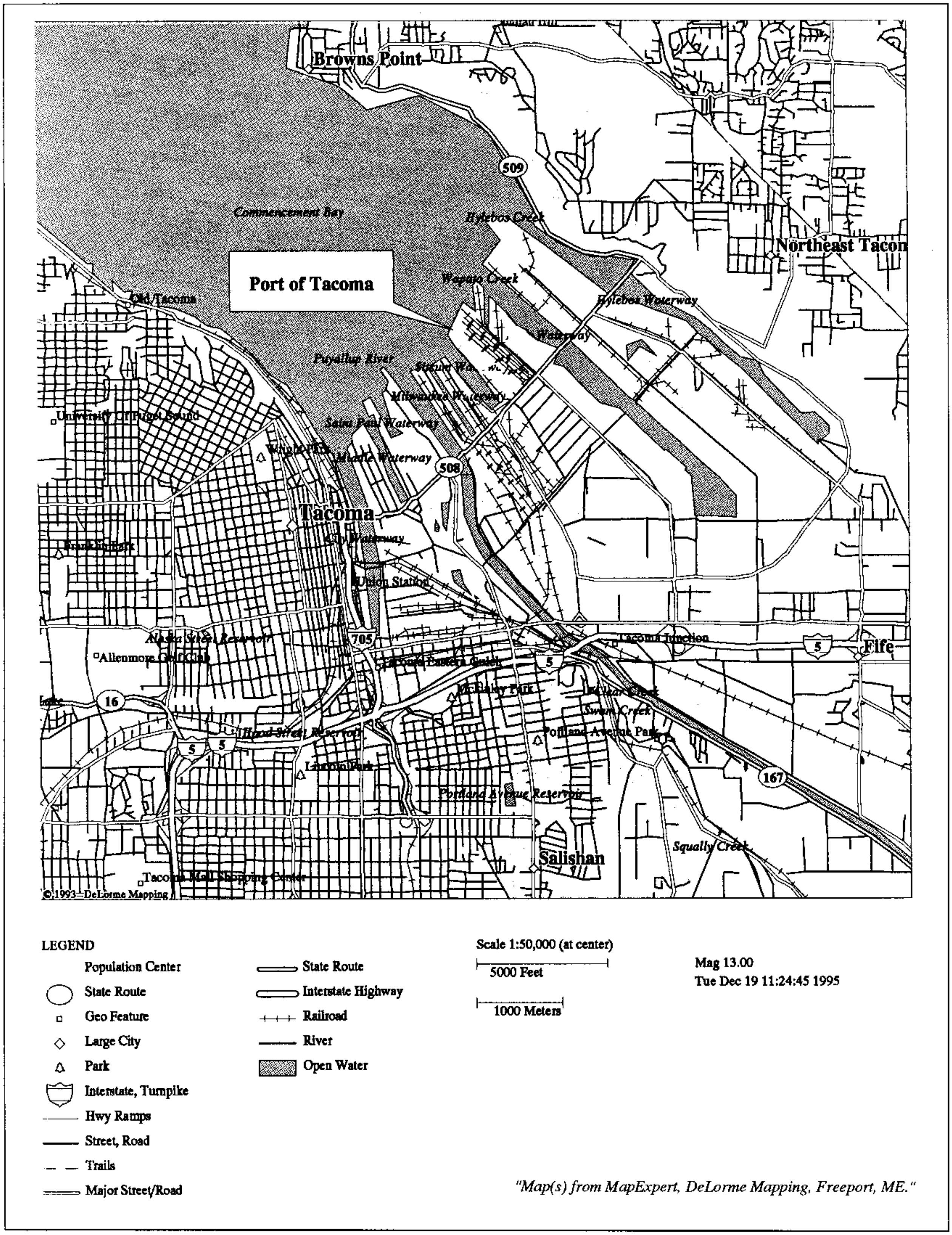


Figure D-30 Map of the Port of Tacoma, WA

The port functions as a special purpose district operation under State enabling legislation and is governed by a Board of Commissioners. The Commission owns and operates several terminals, including container and roll-on/roll-off facilities. Stevedoring is performed by private contractors and/or by ship lines leasing facilities from the port.

Commencement Bay has been designated a "Superfund Site" by the Environmental Protection Agency. However, since the acceptance of spent nuclear fuel through the Port of Tacoma would neither affect the activities being conducted in response to the "Superfund Site" designation, nor would it add any additional burden to this designation, the "Superfund Site" designation has no bearing on the proposed action.

Berths A, B, and C of Terminal 7 are primarily public general cargo facilities handling breakbulk and dry bulk cargoes. Depths alongside range from about 12.2 m to 15.2 m (40 to 50 ft), and it has two 36 metric ton (40 ton) gantry cranes and one 36 metric ton (40 ton) multi-purpose bulk unloading crane. Terminal 7, Berth D (Husky Terminal) is the primary container terminal, and has one 274 m (900 ft) long container berth, 3 container cranes [two 45 metric ton (50 ton) and one 50 metric ton (55 ton)), and 15.2 m (50 ft) of depth alongside at mean low water. It has 14 ha (33 acres) of terminal area with access to the 9,512 m^2 (102,400 ft^2) container freight station and a 8,920 m^2 (96,000 ft^2) transit warehouse located near Berths A and B.

The Husky Terminal is about 4.8 km (3 mi) from the Port of Tacoma road access (Exit 136) to Interstate 5 immediately outside the port complex. While a somewhat longer route, Interstate 5 South connects with I-84 East near Portland, OR, avoiding the added risks of trucking spent nuclear fuel over Snoqualmie Pass to Eastern Washington during the winter. Ship berths are served by the Port Belt Line Railroad, and the port is served by the Burlington Northern and Union Pacific Railroads, which interline with eastern and southern railroads. All Terminal 7 berths are adjacent to the North Intermodal Railroad Yard, which consists of 10.4 ha (26 acres) of yard area and 5,340 m (17,500 ft) of trackage. A second intermodal rail terminal, the South Intermodal Rail Yard, is also located within the port for use by all port shippers (Jane's, 1992; AAPA, 1993; POT, 1994).

Tacoma is served by over a dozen containership and breakbulk ship lines including ELMA, Evergreen, Hyundai, IMT, "K" Line, Maersk, MOL, Navianca, Naviera Pacifico, NOSAC, PCL, Sea-Land, South Pacific Interline, Totem Ocean Trailer Express, Wallenius, and Wallno (Jane's, 1992; AAPA, 1993). These lines provide service with most of the Pacific Rim, including Australia and Japan, and also have service with the Mediterranean (Jane's, 1992; AAPA, 1993).

Other Pertinent Information: According to the port's Director of Risk Management, shipments of spent nuclear fuel could be prohibited by the City of Tacoma Harbormaster's Office, but no formal regulatory restriction was identified. The port has had no identifiable experience with shipment of spent nuclear fuel (SNL, 1994; Paulsen, 1993; NRC, 1993). Security is maintained at Terminal 7 by the port Police around the clock, with locations for segregation and temporary storage of hazardous cargoes (special guards would have to be provided by the shipper for spent nuclear fuel) (Paulsen, 1994). The Tacoma Fire Department provides response for accidents, and the port security personnel are trained in emergency response in cooperation with the Fire Chief (McLendon, 1994). There is also the possibility that the Husky Terminal may begin handling ammonium nitrate in bulk, which (because of the explosion potential) would have to considered in the event the port were to receive spent nuclear fuel shipments (Paulsen, 1994). The U.S. Coast Guard accident statistics for the period 1991-1993 for the Puget Sound indicate a total of 54 reportable accidents (USCG, 1994b). Given the high volume of ship traffic, the accident frequency is considered to be low.

As is the case with Seattle, there is substantial environmental concern about environment damage, and the entire Puget Sound area is subject to severe earthquakes and volcanism. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Tacoma, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a high seismic zone with an acceleration of 0.30 g. There have been two major earthquakes in the Puget Sound area this century; a modified Mercalli intensity VII on April 13, 1949 and a modified Mercalli intensity of VIII on April 29, 1965 (Bolt, 1978). On May 18, 1982, Mount Saint Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascades Range, from Canada to Northern California, are volcanic in origin, and potentially active (Foster, 1971; Hamilton, 1976; IPA, 1993)

The 1990 population within 16 km (10 mi) of the port terminals was 511,575. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 601,000; Oak Ridge Reservation, 431,000; Idaho National Engineering Laboratory, 157,000; Hanford Site, 98,600; and Nevada Test Site, 379,000. Populations along rail routes to four of these sites are slightly larger, but the population along the rail route to Nevada Test Site is slightly smaller (this is largely due to primary use of interstate highways through Salt Lake City and Las Vegas, use of U.S. 95 south from Oregon would reduce the population along truck routes. These populations are shown in Tables D-6 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,720 km (2,931 mi); Oak Ridge Reservation, 4,280 km (2,658 mi); Idaho National Engineering Laboratory, 1,310 km (815 mi); Hanford Site, 399 km (248 mi); and Nevada Test Site, 2,160 km (1,344 mi). Distances along rail routes are longer.

Environmental Conditions

A variety of aquatic species are found in Puget Sound. Several animal species, with special status, may also be found in this area. A variety of marine mammals can be found in the central Puget Sound including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the endangered gray whale, the endangered humpback whale, the threatened Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995). These species are not reported at the port. The U.S. Fish and Wildlife Service reported that the bald eagle and marbled murrelet, both listed protected species, may occur in the vicinity of the port (Frederick, 1994). Bald Eagles can be found throughout this coastal zone, and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The U.S. Fish and Wildlife Service's Ecological Inventory for the Puget Sound area indicates that the habitat of Commencement Bay is used by a variety of birds including: shorebirds, gulls, sandpipers, turnstones, yellowlegs, herons, rails, great blue herons, waterfowls, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American widgeons, pintails, mallards, seabirds, cormorants, alcids, common murre, and the pigeon guillemot. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of Chinook salmon, coho salmon, chum salmon, and pink salmon are found in the Puyallup Waterway/River and use this water body and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, a number of seabird colonies exist along the shoreline of Commencement Bay. Areas of the Puget Sound, north of Commencement Bay, are also used as haulouts by the California sea lion. Areas of estuarine wetlands are located along the northern shore of Commencement Bay (WDW, 1994b).

Commencement Bay has been designated a "Superfund Site" by the Environmental Protection Agency. However, since the acceptance of spent nuclear fuel through the Port of Tacoma would neither affect the activities being conducted in response to the "Superfund Site" designation, nor would it add any additional burden to this designation, the "Superfund Site" designation has no bearing on the proposed action.

Climatic Conditions

See Section D.2.2.21 (Seattle) for climatic information, since conditions in Tacoma, WA are essentially the same.

D.2.1.10 Wilmington, NC

The Port of Wilmington, NC is located on the east bank of the Cape Fear River, about 42 km (26 mi) above its mouth on the Atlantic Ocean. It is the leading port of North Carolina, and its major export is wood pulp. It handles about 110,000 20-ft equivalent units per year, representing about 30 percent of total tonnage. The major terminals are down river from the city. A Federal project maintains a 12.2 m (40 ft) channel over the ocean bar into the Cape Fear River, and then 11.6 m (38 ft) to the port. A new dredging program will deepen the approach channel to 12.2 m (40 ft). The approach to Wilmington, up the Cape Fear River, is more open than many river approaches but has restricted segments. The minimum channel width is about 120 m (400 ft). Currents in the river conform to the channel (DOC, 1993d; FHI, 1993c; NCSPA, 1994). A map of the port is shown in Figure D-31.

The port is owned and operated by the North Carolina State Ports Authority (NCSPA), a State agency. It is a modern container and general cargo facility with over 92,900 m^2 (more than a million ft^2) of covered, sprinklered storage and a total of 11 berths, two of which are open. The port has over 40 ha (100 acres) of paved, open area and 10 ha (25 acres) of semi-improved storage area. The Wilmington wharves are of concrete pile construction, rubber fendered, with a total frontage of about 2,000 m (6,568 ft). Berths 6 to 9 are dedicated containership berths with the remaining berths used for various kinds of general cargo. All of the main cargo berths have a depth alongside at mean low water of 11.6 m (38 ft). The terminal has three, 40.6 metric ton (45 ton) container cranes and two, 50.8 metric ton (56 ton) container cranes, plus three gantry cranes ranging from 40.8 metric ton (45 ton) to 204 metric ton (225 ton) (Jane's, 1992; AAPA, 1994; FHI, 1993c; NCSPA, 1994).

Terminal access via truck is through the controlled South Gate Container Entrance. Truck shipments of spent nuclear fuel from Wilmington to southern destinations are from U.S. Routes 17, 74, 76 and 421 to Interstates 95 and 40. The local routes are accessed about 3 km (2 mi) north of the terminal where they cross the Cape Fear River using the lift bridge. Northern and western long distance routes are via Interstate 40 which connects with State Highway 132 about 16 km (10 mi) north of the city. Wilmington container berths are served shipside by the port rail system and the CSX Railroad. There is also an intermodal trailer-on-flat-car and container-on-flat-car rail yard within the container port. While not currently operational, the port is negotiating with CSX for resumption of intermodal rail service at that facility. At the present time, most rail cargo which requires intermodal connections is trucked to the Charlotte Intermodal Terminal (Wilson, 1995).

The port is serviced by over 30 container lines, including Yang Ming, Polish Ocean, Allied Scandinavian, Central Gulf, Zim, Hanjin and Turkish Cargo Line, plus several regularly scheduled breakbulk shipping companies. These lines provide service from Northern Europe, the Mediterranean, Mideast, East and South Africa, South America, the Far East, Australia, and other shipping centers of the world (Southern Shipper, 1993; Jane's, 1992).

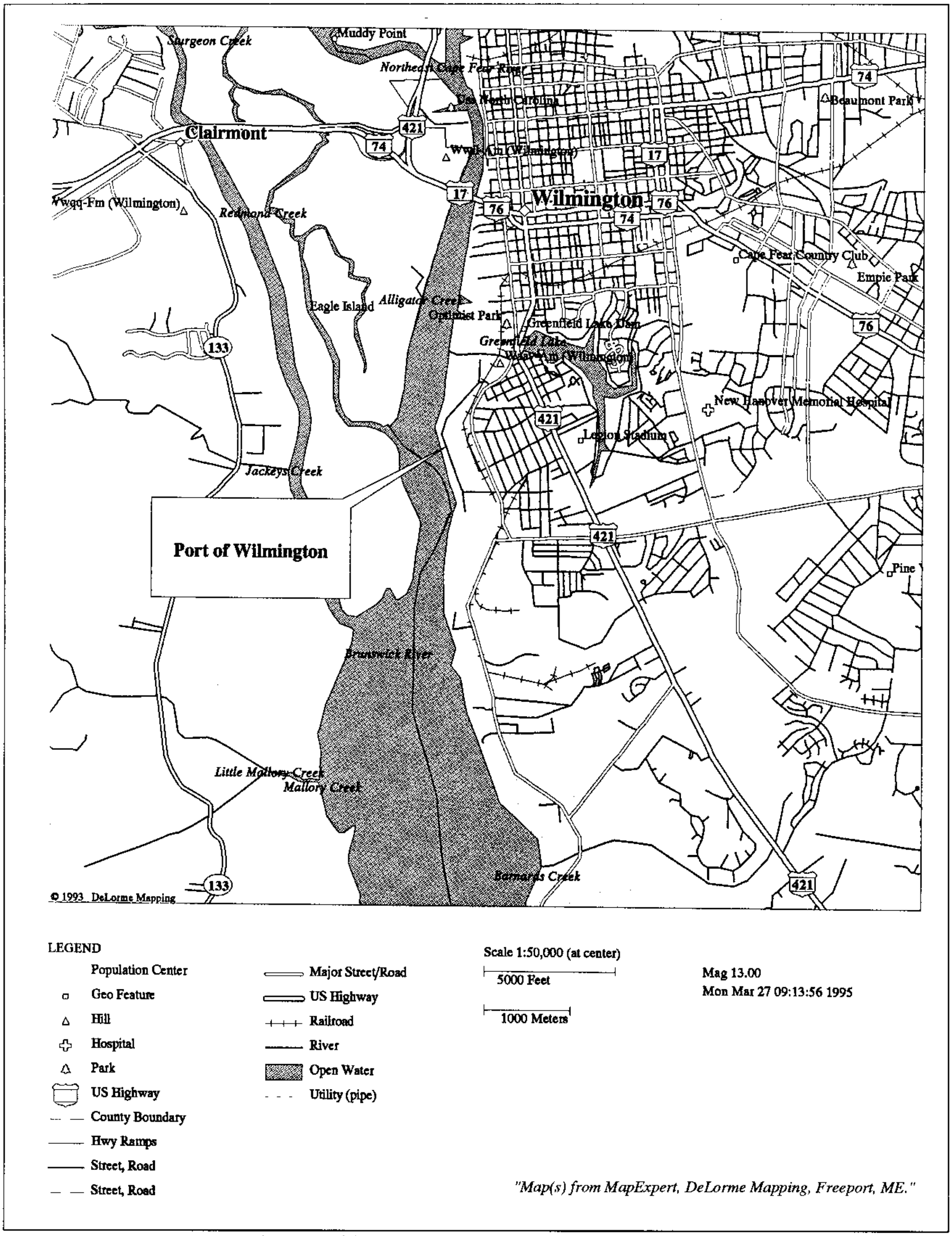


Figure D-31 Map of the Port of Wilmington, NC

Other Pertinent Information: There are no known restrictions on the receipt and handling of spent nuclear fuel through the port, although the Mayor has provided notice that the city is not convinced that the port is desirable for spent nuclear fuel shipments (Betz, 1994). This position was echoed by the Port's Executive Director, who noted that permission to visit the port must come from the State Port Commission, and that the Governor was opposed to handling spent nuclear fuel at State ports (Scott, 1994). Wilmington has handled the import shipments of enriched uranium for nuclear fuel fabrication consigned to a General Electric commercial nuclear fuel fabrication plant north of Wilmington, the exports of the finished nuclear fuel assemblies, and has also handled containerized Class A explosives (Wilson, 1993). The Sandia National Laboratories Radioactive Materials Postnotification Database was queried in April 1994, and the data showed that Wilmington received two shipments of spent nuclear fuel from Japan on February 3, 1986 and transhipped the casks to Savannah River Site the same day (SNL, 1994).

The port is located several miles downstream of the business district in an area of increasing industrial development, although there is some residential housing bordering the complex. The Military Ocean Terminal at Sunny Point is also located on the Cape Fear River, north of Southport, NC, and south of Wilmington, NC.

Port officials are part of an emergency response team headed by the Coast Guard and the Wilmington Fire Department. There are two fire stations within 3 km (2 mi) of the port, with a 5-minute response time (Scott, 1994). All operational personnel working within the terminal, including longshoremen, are given basic hazardous materials training, but training does not deal specifically with spent nuclear fuel.

Security at the port is provided by a 2 m- (6 ft-) high perimeter fencing topped with barbed wire, and a North Carolina State Ports Authority Police Force, which maintains a 24-hour patrol and surveillance. Armed officers are commissioned by the City Police Department, and unarmed guards at the gates are employed by the port (Scott, 1994). A North Carolina State Ports Authority Safety Manager reports to the Director of Operations and is responsible for all safety aspects of the terminal. A tanker terminal and petroleum storage depot are located immediately adjacent downstream of the port. Immediately north of the terminal, on the same side of the river, is an asphalt and chemical storage marine terminal. There is little ship traffic on the River, north or south of the State docks, and therefore there is little conflicting traffic or cargoes.

There are no known environmentally sensitive areas in the immediate vicinity of the terminal, but due to area resorts and recreational activity, there is heightened environmental awareness. The port is subject to hurricanes and tropical storms, as discussed below.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Wilmington, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 115,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 64,700; Oak Ridge Reservation, 128,000; Idaho National Engineering Laboratory, 507,000; Hanford Site, 556,000; and Nevada Test Site, 570,000. Populations along rail routes to these sites are slightly longer. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 500 km (310 mi); Oak Ridge

Reservation, 820 km (509 mi); Idaho National Engineering Laboratory, 4,100 km (2,546 mi); Hanford Site, 4,770 km (2,963 mi); and Nevada Test Site, 4,260 km (2,650 mi). Distances along rail routes are slightly longer for Western sites, but about the same for Eastern sites.

Environmental Conditions

North Carolina has given the lower portion of the Cape Fear River three different stream classifications. From the Northeast Cape Fear River to the confluence with the Cape Fear River the waters are classified as SC-swamp. From the mouth of the Northeast Cape Fear to a point between Snow and Federal Points, the waters are classified as SC. From Snow and Federal Points oceanward the waters are classified as SA. SC waters are tidal waters suitable for fishing, fish and wildlife propagation, secondary recreation and other water uses requiring lower quality. The term "swamp" denotes waters with slow velocity. Class SA waters are suitable for shellfishing, primary recreation, as well as all of the activities approved for Class SC waters (NCDEHNR, 1992). According to the U.S. Fish and Wildlife Service's Ecological Inventory Map for Beaufort, NC, the Port of Wilmington is located in a low salinity estuarine habitat (generally 0.5 to 5 parts per thousand) and tidal freshwater habitat. Below Wilmington at Campbell Island, the river changes to a mid-salinity estuarine habitat (generally 5 to 16.5 ppt). The Cape Fear River near MOTSU changes once again to a high salinity estuarine habitat (generally 16.5 to 30 ppt) (FWS, 1980a).

The lower Cape Fear River supports a large number of aquatic and terrestrial species. There are both invertebrate and fish species of commercial and recreational value found in the Cape Fear River near the Port of Wilmington. Species sought by commercial and recreational fisherman include flounder, trout, spot, croaker, bluefish, Spanish mackerel, and king mackerel. Shellfish sought include penaeid shrimp and blue crabs.

The Natural Heritage Program of the North Carolina Department of Environment, Health, and Natural Resources reports that the area around the state port has not been systematically inventoried for rare species (Smith, 1994). However, DEHNR reports that the lower Cape Fear River, from Wilmington to the mouth of the river at Smith Island, is brackish and contains numerous rare animals. The shortnose sturgeon (State and Federal Endangered Species) rarely occurs in the river, whereas manatees (State and Federal Endangered Species) occasionally occur, especially in the summer. American alligators (State and Federal Threatened Species) can be found in tributary streams. The freckled blenny, spinycheek sleeper, opossum pipefish, and marked goby are other rare marine fishes that inhabit the river.

A large number of aquatic species may be found in the lower Cape Fear River and along the southern coast of North Carolina (Horning, 1994; U.S. Army, 1993; FWS, 1980a). There are many animals with special status in this area, including various types of whales, sea turtles, and birds. State or Federally protected endangered or threatened aquatic species in this area include the shortnose sturgeon (fish), finback whale, Florida manatee, humpback whale, right whale, sei whale, and sperm whale (mammals), Arctic peregrine falcon, bald eagle, piping plover, red-cockaded woodpecker, wood stork (birds), and the American alligator, green sea turtle, hawksbill sea turtle, Kemp's ridley sea turtle, leatherback sea turtle and the loggerhead sea turtle (reptiles and amphibians).

Climatic Conditions

The elevation of this region is approximately 12 m (40 ft) above sea level and is more variable than the coastal plain surrounding the Norfolk, VA, area.

The maritime location of the Wilmington, NC, area makes the climate unusually mild for its northern latitude. All wind directions from the east-northeast through the southwest have some moderating effect on the local climate, due to the relatively warm and cool ocean in the winter and summer seasons, respectively. The area rarely experiences cold episodes where the temperature falls below -18°C (0°F). However, cold air outbreaks do occur, causing sharp fluctuations in winter temperatures. Rainfall in the area is generally considered ample and evenly distributed throughout the year, with the bulk of the precipitation occurring during the summer months. The bulk of this rainfall is generally associated with afternoon and evening thunderstorms. In contrast, the winter rains tend to be of the slow, steady type lasting, generally, one to two days. As is common at Atlantic coastal localities at this latitude, the late summer and early fall months bring the possibility of hurricanes and tropical storms to the Wilmington, NC area. These storms are capable of generating high winds, above normal tides, and torrential rains. The latter two are also capable of creating widespread local flooding of low-lying coastal areas (NOAA, 1992b).

D.2.2 Other U.S. Ports Meeting the Appropriate Experience Criteria

The ports described in this section are those that were initially identified as acceptable based on experience with containerized cargo, but that were subsequently dropped from further consideration based on evaluation against other criteria. Those criteria and the evaluation process are described in Section D.1.

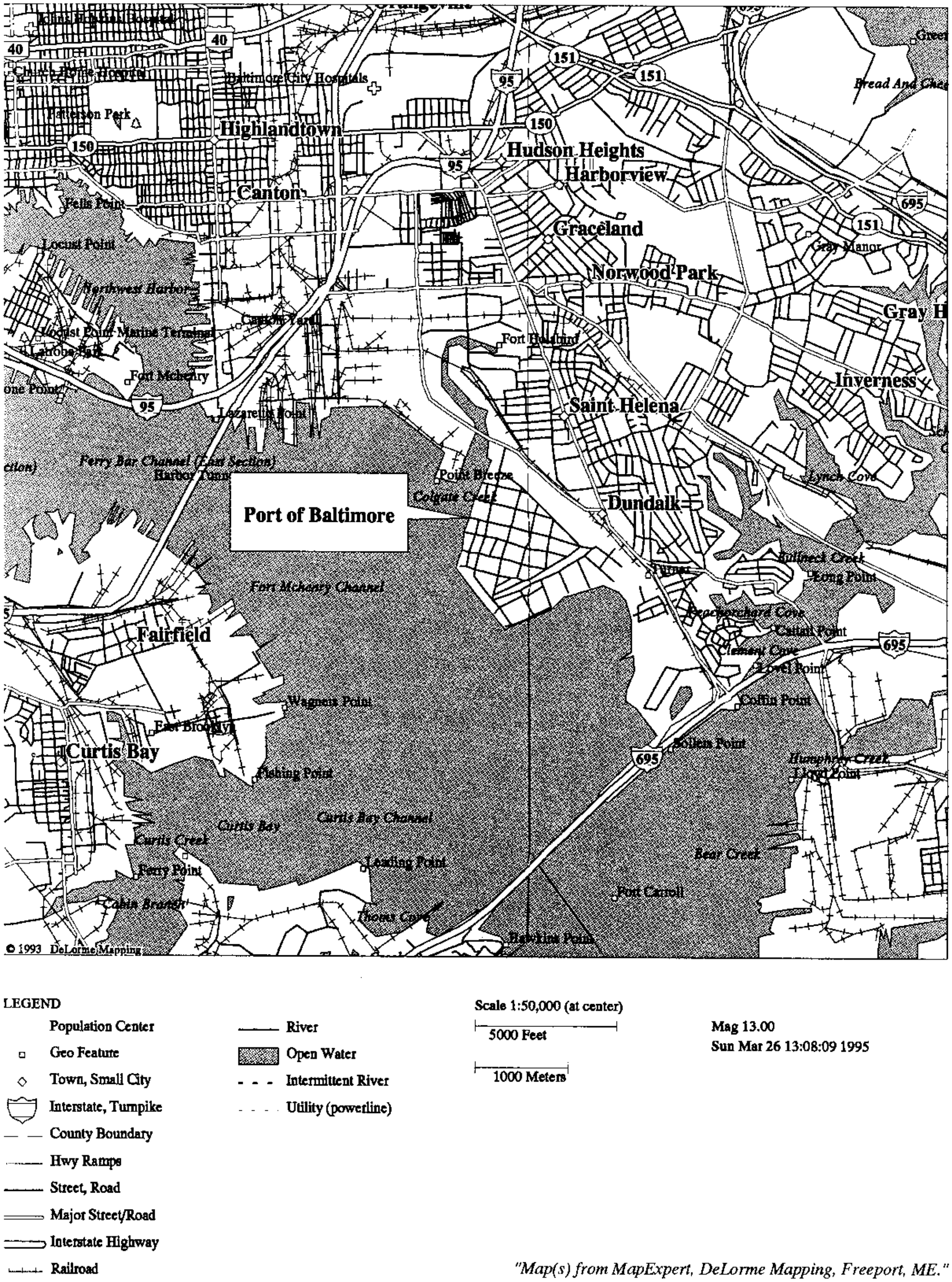
D.2.2.1 Baltimore, MD

The Port of Baltimore, one of the major ports of the United States, is established on the upper Chesapeake Bay at the head of tidewater navigation on the Patapsco River. It is situated 13 km (8 mi) from the entrance to the Patapsco, 240 km (150 mi) from the Virginia Capes, and 1670 km (104 mi) from the Delaware Capes.

Depths within the harbor range from 15.2 m (50 ft) to 12.2 m (40 ft). Federal project depths are 15.2 m (50 ft) in the main channel between the Virginia Capes and Fort McHenry in the Baltimore Harbor. Access to the port can be gained via the Delaware Bay and River, and the Chesapeake and Delaware Canal, although this route is not recommended due to the restrictive nature of the transit. The preferred route is via the Virginia Capes and Chesapeake Bay. The Chesapeake Bay is considered part of the transit to Baltimore; this includes 240 km (150 mi) up the Bay under the Chesapeake Bay Bridge to the Patapsco River (DOC, 1993c). A map of the Port of Baltimore is shown in Figure D-32.

In the Port of Baltimore, the Maryland Port Administration (MPA) owns and operates the two principal general cargo terminals — Seagirt and Dundalk Marine Terminals. Together these terminals provide a total of 338 ha (835 acres) of cargo handling space on deepwater channels. Both terminals are ports of call for many of the world's largest container and roll-on/roll-off shipping lines. Seagirt is capable of handling (length and draft) the largest post-Panamax vessels currently in service (Jane's, 1992; AAPA, 1993; D&B, 1993).

Seagirt Terminal: Seagirt is MPA's newest and most modern container facility with three container ship berths, eight cranes, and 107 ha (265 acres) of developed terminal area. Seagirt also has a deeper water depth alongside berths; the three container berths have a depth alongside at mean low water of 12.8 m (42 ft). The terminal has two 313 m each (1,020 ft each) marginal wharves and one 326 m (1,070 ft) marginal wharf. Crane capacity at the terminal consists of four 50.8 metric ton (56 ton) single hoist container cranes, three 50.8 metric ton (56 ton) double hoist container cranes, and one 27.9 metric ton (30.7 ton) single hoist container crane (AAPA, 1993 and 1994; Jane's, 1992; D&B, 1993).



"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-32 Map of the Port of Baltimore, MD

Dundalk Marine Terminal: Dundalk Marine Terminal, located adjacent and eastward of Seagirt, has approximately 231 ha (570 acres) of terminal area and is a combination container, roll-on/roll-off, and breakbulk handling facility. The terminal has 13 barge and ship berths, 11 cranes, and covered storage shed space of more than 37,000 m^2 (400,000 ft^2). Marginal wharves consist of three 808 m total (2,650 ft), two 553 m total (1,820 ft) container berths, and one 305 m (1,000 ft) container berth. Containership berths have a depth alongside of 11.5 m (38 ft). Crane capacity at the terminal includes nine 40.6 metric ton (45 ton) single hoist container cranes and two 61 metric ton (67 ton) gantry whirley cranes (Jane's, 1992; D&B, 1993; AAPA, 1993 and 1994).

The Seagirt and Dundalk Terminals are located in the Dundalk section of the City of Baltimore, east and south of the central business district. The access road to both terminals is bordered primarily by heavy industrial types of businesses with relatively good interstate highway connections. Southbound, the distance from Seagirt to I-695 is roughly 4 km (2.5 mi). The entrance to the Seagirt Marine Terminal is approximately 1.6 km (1 mi) from I-95 connected by Bruening Highway, an industrial roadway that also serves as the main truck access to both terminals. Access to other major interstate highways is via the I-695 Beltway, which would be used to bypass harbor tunnels for Savannah River Site or other southern destinations. Routing and connect time for Dundalk traffic would be virtually the same due to proximity of location to the Seagirt terminal. Seagirt is served by the CSX Railroad, which operates a 16.2 ha (70 acre) intermodal container transfer facility inside the terminal and within 0.3 km (1000 ft) of the ship berths. Conrail serves the Dundalk Terminal for breakbulk cargoes (D&B, 1993; AAPA, 1993 and 1994).

Other Pertinent Information: Security of both terminals is maintained by the MPA Port Police and is deemed to be excellent. There are secure areas for temporary segregation and storage of containers if necessary.

There are no port restrictions against handling spent nuclear fuel. A port safety officer stated that spent nuclear fuel shipments go out of the port with an armed escort (normally at night), and that the port also handles casks (cylinders) of uranium hexafluoride (UF_6) shipments quite frequently. Although there are no known conflicts with other hazardous materials in the immediate terminal area, there is a diversity of marine terminals and ship traffic activity on the Patapsco River which are not deemed to represent a major hazard factor.

There are no known special environmental issues with regards to handling spent nuclear fuel at Baltimore. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Baltimore, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The MPA relies on the hazardous materials teams of the Baltimore City and County fire departments as well as the Coast Guard for response to hazardous materials accidents. The Maryland Department of the Environment also has input on hazardous materials problems. The MPA has an ongoing hazardous materials training program for all port operating personnel, including the longshoremen. Instruction includes dealing with hazardous wastes (but not spent nuclear fuel in particular) in the soil and groundwater due to the former use of the port site.

The 1990 population within 16 km (10 mi) of the port terminals was 1,182,024. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 308,000; Oak Ridge Reservation, 246,000; Idaho National Engineering Laboratory, 482,000; Hanford Site, 531,000; and Nevada Test Site, 665,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,020 km (636 mi); Oak Ridge Reservation, 925 km (575 mi); Idaho National Engineering Laboratory, 3,790 km (2,354 mi); Hanford Site, 4,460 km (2,770 mi); and Nevada Test Site, 4,060 km (2,526 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Gunpowder Falls State Park is located approximately 22 km (13 mi) northeast of the port. The Remington Farms Wildlife Reserve, on the Eastern Shore of Maryland, is approximately 35 km (23 mi) east of the port. The Eastern Neck National Wildlife Refuge is located in the Chesapeake Bay, about 38 km (25 mi) southeast of the port area. The Fort McHenry National Monument and Historic Shrine is located on a point of land approximately 4 km (2 mi) west of the Baltimore port. Numerous State Parks and other wildlife refuges are located along the passageway in the Chesapeake Bay south of the port.

The endangered peregrine falcon occurs in the vicinity of the Port of Baltimore (Wolflin, 1994). These birds feed, in part, on shorebirds and other waterbirds using the waters of the Port of Baltimore. The endangered Delmarva fox squirrel and the great blue heron (State-protected) nest on the Eastern Neck Island (FWS, 1980g). The bald eagle (endangered) also nests in the Eastern Neck Island area.

The Bay contains many beds of commercially valuable oysters and soft-shelled clams. Blue crabs are harvested extensively throughout the Bay area. Commercial harvesting of channel catfish and menhaden also is important in the Bay area. Numerous types of fish use the Bay area, including the waters around the port, for nursery areas. Common fish species include the American eel, blueback herring, hickory shad, alewife, gizzard shad, perch, striped bass, drum, flounder, and others. Sport fishing for these fish is also common. State-protected species include the Atlantic sturgeon and American shad (FWS, 1980g). The western bank of the Eastern Shore is a migratory area for the dabbling duck (nonendangered) and a heavily used migration pathway for geese.

Climatic Conditions

Baltimore is in a region about midway between the rigorous climates of the North and the mild climates of the South and adjacent to the modifying influences of the Chesapeake Bay and Atlantic Ocean to the east and the Appalachian Mountains to the west. The net effect is to produce a more equable climate compared to inland locations of the same latitude.

Rainfall distribution throughout the year is rather uniform; however, the greatest intensities are confined to the summer and early fall, the season for hurricanes and severe thunderstorms. Rainfall during this period occurs principally in the form of thundershowers, and rainfall totals during these months vary appreciably, depending on the number of thundershowers that occur largely by chance in a given locality. Hurricane-force winds, however, may occur on rare occasions due to a severe cold front or a severe thunderstorm. The greatest damage by hurricanes is that produced along waterfronts and shores by the high tides and waves.

In summer, the area is under the influence of the large semipermanent high-pressure system commonly known as the Bermuda High and centered over the Atlantic Ocean near latitude 30°N. This high-pressure system brings a circulation of warm, humid air masses over the area from the deep South. The proximity of large water areas and the inflow of southerly winds contribute to high relative humidities during much of the year.

January is the coldest month, and July the warmest. Winter and spring have the highest average windspeeds. Snowfall occurs on about 25 days per year on the average; however, an average of only 9 days annually produce snowfalls greater than 1.0 in. Although heaviest amounts of snow generally fall in February, occasional heavy falls occur as late as March. Records for the period August 1950 through December 1967 indicate that the average date of the last temperature as low as 32° in the spring is April 15, while the average date of the first temperature as low as 32° in the autumn is October 26 (NOAA, 1993a).

Glaze or freezing rain occurs on an average of two to three times per year, generally in January or February, although some occurrences have been noted in November and December. Some years pass without the occurrence of freezing rain, while in others it occurs on as many as eight to ten days. Sleet is observed on about five days annually. The sleet season begins as early as November in some years, and ends as late as March in some cases, with the greatest frequency of occurrence in January (DOC, 1993c).

D.2.2.2 Boston, MA

The Port of Boston is located on Massachusetts Bay about 93 km (50 mi) west of Cape Cod and is the largest seaport in New England. Boston North Channel is the main entrance to Boston Harbor and Boston South Channel and The Narrows are alternative entrances. A Federal project on the North Channel (to the Mystic River) provides for a channel width of 460 m (1,500 ft) and a depth of 12.2 m (40 ft) in the eastern section, and a width of 270 m (900 ft) and depth of 10.7 m (35 ft) in the western section (DOC, 1993a).

Although there are many obstructions in the Harbor approaches, they are marked by a number of powerful lights, and the principal dangers are buoyed. Because of the heavy traffic to the Harbor, there is a traffic separation scheme extending over 160 km (100 mi) out to sea (DOC, 1993a).

The Massachusetts Port Authority (Massport) is a quasi-governmental authority created by the State Legislature in 1956. The Maritime Division is responsible for the operation, development, and maintenance of the port's three public terminals, including two container terminals (Moran and Conley) and one general cargo facility (Harbor Gateway Terminal) (POB, 1993). A map of this port is provided in Figure D-33.

Moran Terminal: Moran Terminal is located about 7.2 km (4.5 mi) upstream of the Inner Harbor Entrance, on the left side of the ascending bank of the Mystic River in Boston's Charlestown section. It is the largest container terminal in New England and is operated by Massport. The facility consists of 20.2 ha (50 acres) of open storage space, storage capacity for 4,000 20-ft equivalent units, and two container cranes [46 and 71 metric tons (51 and 78 tons)]. It has 335 m (1,100 ft) of marginal wharf and depth alongside of 12.2 m (40 ft). Vessels are limited by the 41.2 m (135 ft) clearance under the Tobin Memorial Bridge (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993).

Moran Terminal is situated about 1.6 km (1 mi) from the intersection with I-93 with access via city streets through the densely populated Charlestown area. The terminal is served by the Boston & Maine Railroad, whose tracks enter the terminal and extend to the pier apron.

Conley Terminal: The Conley Terminal on Castle Island is less than 6.4 km (4 mi) from the designated entrance to Boston Harbor. The northern approach to the terminal is obstructed by islands and shoals that extend 6.4 km (4 mi) from the entrance for a combined distance of about 13 km (8 mi). It is located at the entrance of the Inner Harbor on the South Boston waterfront. The Terminal is operated by a subcontractor to Massport. It has 305 m (1,000 ft) of marginal wharf, and consists of Berths 11-15, and Berth 17. The depth alongside is 12.2 m (40 ft).

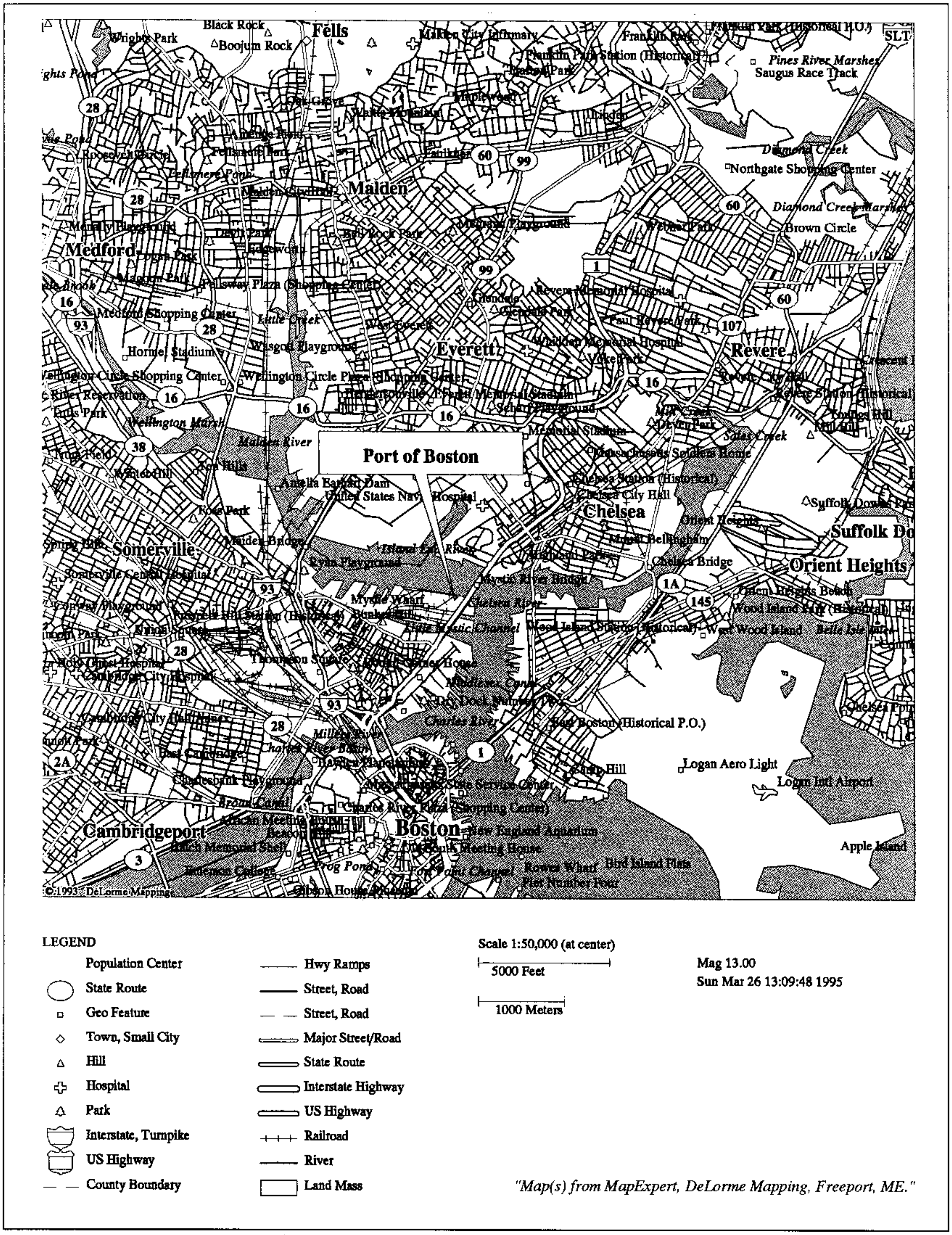


Figure D-33 Map of the Port of Boston, MA

The container terminal, Berth 11, has two 41 metric ton (45 ton) container cranes and an open storage area of 4 ha (9.9 acres). Berth 12 is presently undergoing a $50 million improvement program (to be completed in 1995), and Berths 13-15 are leased to automobile importers. Berths 16 and 17 are served by one container crane (31 metric ton) and are also leased by automobile importers (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993). The terminal is approximately 3.2 km (2 mi) from Route I-93, which is part of the Greater Boston Beltway, which then connects with I-95 and I-90 (the Massachusetts' Turnpike). Access to the terminal is via East and West Broadway, a busy South Boston thoroughfare running through an area of primarily small businesses with some old residential housing. Construction of the Third Harbor Tunnel/Seaport Access Road began in 1992 for better interstate access. The terminal is served by Conrail whose tracks are located outside and at the rear of the terminal.

Massport Marine Terminal: This is a 16 ha (40 acre) facility used for cruise ships and the discharge of automobiles (roll-on/roll-off) and bulk cargo. This terminal is about 1.6 km (1 mi) from I-93 via Northern Avenue (a truck route to the Boston Fish Pier) and other industrial users along the waterfront (Jane's, 1992; AAPA, 1993 and 1994; D&B, 1993; POB, 1993).

Other Pertinent Information: Massport has its own security force, which has police powers at State-owned terminals. Although there is no officially designated space for segregating hazardous materials, the port would provide one if necessary. There are no known regulatory restrictions against handling of spent nuclear fuel at Massport terminals; the Deputy Port Director for Operations did not know if the port has ever handled spent nuclear fuel (Moriconi, 1993).

Massport relies on its fire department, which also has a fireboat, for emergency response for hazardous materials accidents, and on Coast Guard supervision. The port also coordinates its activities with State hazardous materials safety personnel. Massport has a training program for terminal workers at Moran, and recently began an introductory course for longshoremen. Training at leased facilities, like Moran Terminal, is the responsibility of the terminal operator (Moriconi, 1993).

Moran Container is located in the densely populated Charlestown area on the Mystic River across from petroleum and natural gas terminals, and a residential condominium/marina complex. Conley Terminal is in an industrial area with less conflicting use, but access is through South Boston, also a densely populated commercial/residential area.

There are no known special environmentally sensitive areas within the port. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Boston, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (85 mph). The port is located in a moderate seismic zone with an acceleration of 0.30 g.

The 1990 population within 16 km (10 mi) of the port terminals was 1,466,233. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 912,000; Idaho National Engineering Laboratory, 716,000; Hanford Site, 785,000; and Nevada Test Site, 796,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,734 km (1,079 mi); Oak Ridge Reservation, 1,600 km (995 mi); Idaho National Engineering Laboratory, 4,180 km (2,600 mi); Hanford Site, 4,850 km (3,016 mi); and Nevada Test Site, 4,560 km (2,832 mi). Distances along rail routes are about the same for Hanford Site and Idaho National Engineering Laboratory, and are slightly longer for Savannah River Site, Oak Ridge Reservation, and Nevada Test Site.

Climatic Conditions

Three important influences are responsible for the main features of Boston's climate (DOC, 1993a). First, the latitude 42°N places the city in the zone of prevailing west to east atmospheric flow, which encompasses the northward and southward movements of large bodies of air from tropical and polar regions. This results in variety and changeability of the weather elements. Secondly, Boston is situated on or near several tracks frequently followed by systems of low air pressure. The consequent fluctuations from fair to cloudy or stormy conditions reinforce the influence of the first factor, while also ensuring a rather dependable precipitation supply. The third factor, Boston's east coast location, is a moderating factor affecting temperature extremes of winter and summer.

Hot summer afternoons are frequently relieved by the locally celebrated "sea-breeze," as airflows inland from the cool water surface to displace the warm westerly current. This refreshing east wind is more commonly experienced along the shore than in the interior of the city or the western suburbs. In winter, under appropriate conditions, the severity of cold waves is reduced by the nearness of the then relatively warm water. The average date of the last occurrence of freezing temperature in spring is April 8; the latest is May 3, 1874 and 1882. The average date of the first occurrence of freezing temperature in autumn is November 7; the earliest on record is October 5, 1881. In suburban areas, especially away from the coast, these dates are later in spring and earlier in autumn by up to one month in the more susceptible localities.

Boston has no dry season. For most years the longest run of days with no measurable precipitation does not extend much more than two weeks. This "dry spell" may occur at any time of year.

Much of the rainfall from June to September comes from showers and thunderstorms. During the rest of the year, low-pressure systems pass more or less regularly and produce precipitation on an average of roughly one day in three. Coastal storms, or "northeasters," are prolific producers of rain and snow. The main snow season extends from December through March. The average number of days with four inches or more of snowfall is four per season, and days with seven inches or more come about twice per season. Periods when the ground is bare or nearly bare of snow may occur at any time in the winter.

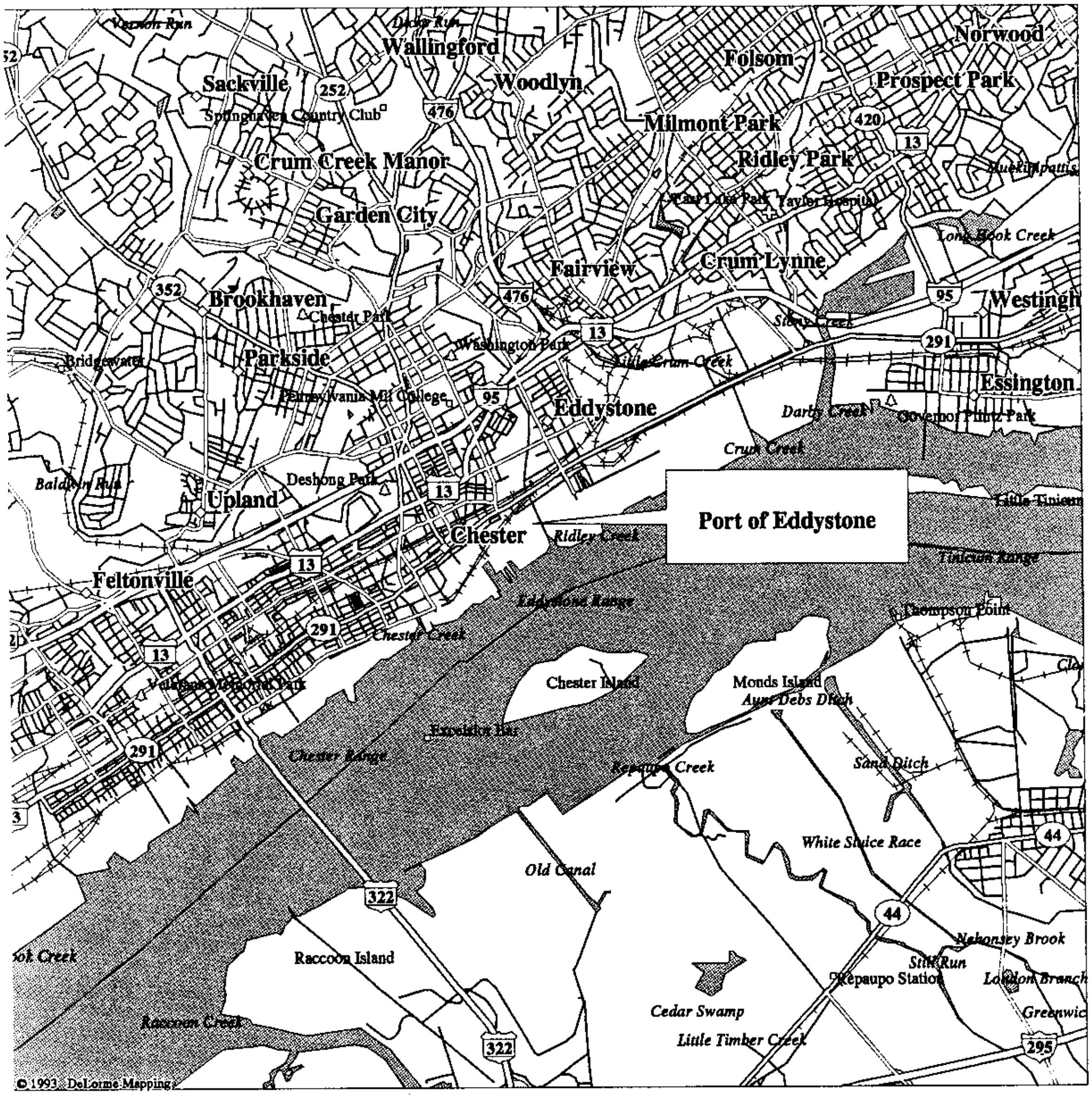
Relative humidity has been known to fall as low as five percent (May 10, 1962), but such desert dryness is very rare. Heavy fog occurs on an average of about two days per month, with its prevalence increasing eastward from the interior of Boston Bay to the open waters beyond. Winds from the east to southwest bring fog and westerly and northerly winds clear the fog away.

At all seasons the heaviest gales are usually from the northeastward or eastward. Although winds of 32 mph or higher may be expected on at least 1 day in every month of the year, gales are both more common and more severe in winter (DOC, 1993a).

D.2.2.3 Eddystone, PA

The location of Penn Terminals at Eddystone is on the former site of Pennsylvania Shipbuilding Company's North Yard Wharf, just upstream of the entrance to Ridley Creek in Eddystone, PA. It is located approximately 18 km (11 mi) south of Philadelphia. It is approximately 1.5 km (1 mi) above the small port of Chester, PA. Penn Terminals, Inc. is one of several independent Delaware River port terminal operators who come under the marketing umbrella of the Delaware River Port Authority (AAPA, 1994). A map of the port is shown in Figure D-34.

Geophysical and navigational data for Penn Terminals is, with the exception of the approach to the Terminal, the same as for Philadelphia and other Delaware River Ports in the immediate vicinity (AAPA, 1994).



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Park
- Interstate, Turnpike
- US Highway
- County Boundary
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- US Highway
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Sun Mar 26 13:11:43 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-34 Map of the Port of Eddystone, PA

Penn Terminals was founded in 1986 to manage containerized cargoes but subsequently expanded its scope of services to include breakbulk and project cargoes. Penn Terminals' brochure states that they handle about 250 ship calls a year (PT, 1994). A port official reported that the Terminal handles 30,000 to 50,000 20-ft equivalent units a year, including some hazardous and radioactive materials (Davis, 1994). According to the Sandia National Laboratory's Radioactive Materials Postnotification (RAMPOST) Database, on April 17, 1991, this port was used for receipt of about 1.4×10^{16} Bq [366,000 curies (Ci)] of cobalt-60 for shipment to Dickerson, MD, in a Type B cask comparable to those used for spent nuclear fuel shipments (SNL, 1994). There was no indication of foreign research reactor spent nuclear fuel receipts since October 1984, when the database was established.

The Terminal features 40.7 ha (71 acres) of storage area, including 23,200 m^2 (250,000 ft^2) of covered storage. The terminal has 335 m (1,100 ft) of marginal wharf, container gantry cranes, a 27 metric ton (30 ton) and a 41 metric ton (45 ton) and a heavy lift truck crane with a capacity of 220 metric tons (240 tons). Rail service is provided by Conrail. Access to Interstate 95 is about 1.6 km (1.0 mi) from the terminal via industrial and old residential streets (PT, 1994; AAPA, 1994).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Eddystone, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 609,241. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 400,000; Oak Ridge Reservation, 300,000; Idaho National Engineering Laboratory, 600,000; Hanford Site, 600,000; and Nevada Test Site, 700,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,170 km (725 mi); Oak Ridge Reservation, 1,060 km (660 mi); Idaho National Engineering Laboratory, 3,930 km (2,440 mi); Hanford Site, 4,590 km (2,850 mi); and Nevada Test Site, 4,180 km (2,600 mi). Distances along rail routes are about the same.

Environmental Conditions

Monds Island and Chester Island are directly across the shipping channel from the port, and Little Tinicum Island is 1.8 km (1.1 mi) upriver. Tidal flats surround these islands, which are comprised of marshes and wetlands. The Tinicum National Environmental Center, located approximately 3.5 km (2.2 mi) to the northeast on Darby Creek, is a nationally recognized wetlands and environmental education center.

The Port of Eddystone is located within Zone 4 (tidal river) of the Delaware River. Protected water uses for Zone 4, which encompasses River Miles (RM) 79-95, are water supply (industry), wildlife, resident fish maintenance, anadromous fish passage, secondary contact, and navigation (DRBC, 1994). However, several uses within Zone 4 are currently impacted, including: fish and other aquatic life due to low dissolved oxygen levels from point source discharges, and fish and shellfish consumption due to chlordane and PCB contamination from point and nonpoint source discharges.

The Delaware River at Eddystone is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms that are typically found in the waters of this area include: American shad, Atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel

(FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. The Water Quality Section of the Pennsylvania Department of Environmental Resources reported that 67 species of fish are full or part-time residents of this part of the Delaware estuary (Boyer, 1994). Most importantly, the area of the river between Monds Island, Chester Island, and Little Tinicum Island and the islands' backwaters, is an important spawning site for the striped bass.

This area of the Delaware River serves as a sport fishery for striped bass, American shad, blue-claw crabs, white perch, largemouth bass, and catfish. There is also limited commercial fishing for American eels and American shad. There is only low to medium recreational use of this part of the Delaware River due to the high volume of tanker and freighter traffic (Boyer, 1994).

The U.S. Fish and Wildlife Service reported that except for occasional transient species, no Federally listed or proposed threatened or endangered species under their jurisdiction are known to exist in the port's impact area (Perry, 1994). Similarly, the Pennsylvania Natural Diversity Inventory reported that it did not expect any impact on rare, threatened, or endangered plant species in this location (PNDI, 1994).

Climatic Conditions

The climate of the Eddystone region is similar to that of Philadelphia, PA. The area is moderated by the Appalachian Mountains to the west and the Atlantic Ocean to the east. These geographic features cause periods of extreme temperatures to be short-lived in this region (generally, four days). On occasion during the summer months, the area is dominated by maritime tropical air masses, which contribute to elevated local temperature and humidity levels. The average annual precipitation of 105.2 cm (41.42 in) is relatively evenly distributed throughout the year, with maximum amounts occurring during the late summer months. The summer precipitation regime is dominated by localized thunderstorms and are subject to the influence of the urban heat island effect and local topography, which create varying rainfall amounts across the city for an individual event. Singular snowfall events that generate accumulated totals of greater than 25.4 cm (10 in) have a 5-year recurrence interval on average. The prevailing wind direction has a bimodal distribution, being southwesterly during summer and northwesterly in the winter months. The annualized average prevailing wind direction is from the west-southwest. Due to this region's inland location, destructive winds are comparatively rare from tropical cyclones and tornadoes. High winds are generally associated with frontal passages/low pressure systems and thunderstorms in the winter and summer months, respectively. However, flooding on the Schuylkill River normally occurs twice annually, usually associated with strong thunderstorms, with the duration of these events generally lasting less than 12 hrs. The Delaware River is rarely observed at or above flood stage (NOAA, 1992h).

D.2.2.4 Elizabeth, NJ

New York Harbor is the principal entrance by water to New York City and the surrounding ports. The harbor is divided by the Verrazano Narrows into the Lower Bay and Upper Bay. Using the Verrazano Narrows Bridge as a reference point, Port Elizabeth is approximately 18 km (11 mi) from the Lower Bay and the Atlantic Ocean via Kill Van Kull. The Battery, the southern tip of Manhattan, is at the junction of the East River and Hudson River. New York Harbor includes New York City, Staten Island, and the New Jersey principal ports of Perth Amboy, Port Elizabeth, Port Newark, and Bayonne. The project depth of the channels leading from the sea through the Lower Bay, Narrows and Upper Bay is 13.7 m (45 ft). Depths in the Kill Van Kull leading to the New Jersey container terminals is 10.7 m (35 ft). The approaches to New York Harbor are open, but highly trafficked. The 13 km (8 mi) down the Kill Van Kull to Port Elizabeth is restricted (DOC, 1993b). A map of the port is provided in Figure D-35.

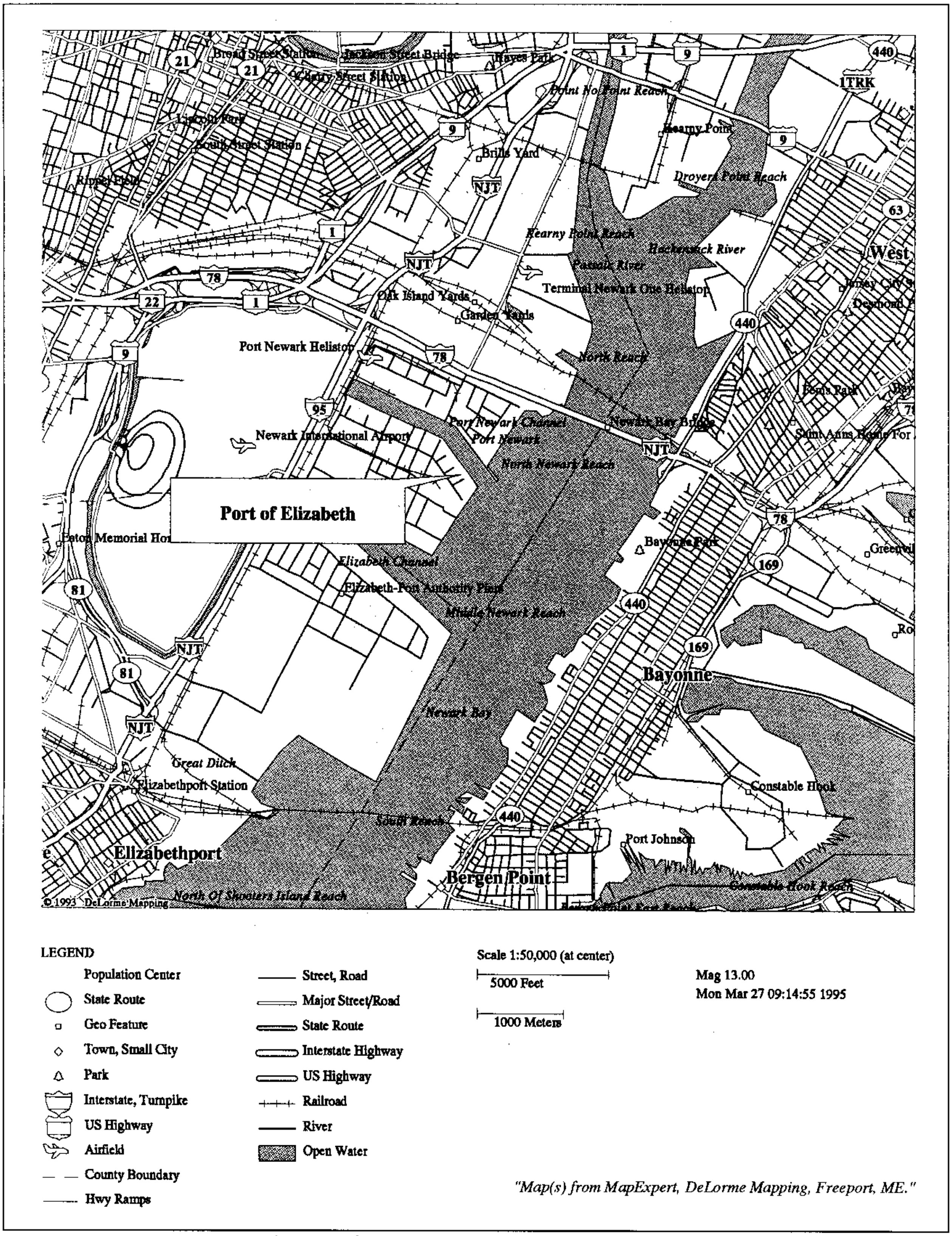


Figure D-35 Map of the Port of Elizabeth, NJ

Unlike many ports of the world, no single governmental or public agency in New York is responsible for controlling the overall operation of the port. Port administration is divided among many organizations, both private and public, which have an institutional interest in port activities.

The Port Authority of New York and New Jersey (Port Authority) is a quasi-public agency established in 1921 by treaty between the states of New York and New Jersey to deal with the planning and development of terminal and transport facilities, and to improve and protect the commerce of the port district.

The Port Authority's main maritime facilities are located in Elizabeth, Port Newark, and Hoboken, New Jersey, and in New York, at Erie Basin and Columbia Street terminals in Brooklyn. The City of New York owns the South Brooklyn Marine Terminal, Red Hook Marine Terminal (also in Brooklyn), and Howland Hook Marine Terminal on Staten Island. The latter is a major container terminal now leased to the Port Authority. Global Terminal, a privately owned and operated container facility, is located in Jersey City. All told, there are five separate container areas within the harbor equipped with a total of 35 container cranes along a total quay length of 8,000 m (approximately 5 mi), and a total berth area of about 500 ha (1,236 acres) (Jane's, 1992; AAPA, 1993).

With the exception of Global Terminal, all of the foregoing terminals are leased from the Port Authority or the City of New York and operated by terminal operating companies or steamship lines. Since virtually any one of these terminals would be physically capable of handling containers of spent nuclear fuel, description of the port's capabilities is limited to a single terminal, the Port Authority Marine Terminal, within the Port Authority's Port Elizabeth/Port Newark container complex. The Port Elizabeth/Newark area has direct access to the New Jersey Turnpike and is farthest removed (relatively) from centers of population.

Sea-Land Terminal (Elizabeth NJ): Berths 88-98 on the southeast corner of Elizabeth Channel have 1,403 m (4,603 ft) of marginal wharf. The terminal has 12.2 m (40 ft) depth alongside at mean low water. Sea-Land has crane capacities of six 40.6 metric ton (44.8 ton) container gantry cranes. Truck access to the New Jersey Turnpike (I-95) is via Port Newark (Exit 14) or Exit 13A in Elizabeth. The latter is reached via McLester Street to State Route 81 to the Turnpike. The route is almost entirely within the Port Authority Marine Terminal complex and distance traveled is estimated to be about 4.8 to 6.4 km (3 to 4 mi), respectively. The Sea-Land Terminal is adjacent to the Conrail Portside and the Port Authority intermodal rail yards.

In addition to Sea-Land, the Terminal is used by Hanjin Shipping Lines, Ltd., Italia Line, Nedlloyd, P & O Containers, Samskip, S.C.I. Line, Spanish Line, and Transroll Navegacao, SA. The list of container lines calling at other terminals is extensive and represents the major container carriers of the world (Jane's, 1992; AAPA, 1993).

Other Pertinent Information: Individual Terminals are responsible for their own security arrangements. However, it is believed that the New York Port Association controls and may serve as watchmen. All terminals are fenced with controlled access and 24-hour surveillance. A port official did not know what type of short-term storage arrangements exist at the Sea-land Terminal; however, he believed there is provision for segregating hazardous cargoes. He also did not know if there are any restrictive ordinances pertaining to spent nuclear fuel or if the port has handled it (Hennessy, 1993). Available data indicates spent nuclear fuel shipments have not been handled at least since 1979 (NRC, 1993; SNL, 1994).

The Kill Van Kull waterway, serving Port Elizabeth/Newark terminals, is also the approach route to the refineries and petroleum storage depots located along the Arthur Kill to the south. There is a great diversity of traffic and cargoes in the harbor but, due to the layout of the terminals, this diversity and traffic are not considered a major concern.

The Coast Guard and fire departments from the cities of Elizabeth and Newark, respond to hazardous materials incidents within terminals located within their municipalities. The Union County hazardous materials team responds to accidents in Port Elizabeth, and the Newark hazardous materials team in Port Newark. Sea-Land and other terminal operators have contracts with private companies for oil and chemical spill cleanup and/or decontamination work. It is not known what type of hazardous materials training is provided by terminal operators and/or the Port Authority (Hennessy, 1993). Training normally is provided in such large port operations. This was not investigated further because the port was not included in the final list selected for detailed assessment due to the extremely large populations around the port.

The Port Elizabeth/Newark terminals are separated from the urban city centers bearing their names. However, both are adjacent to Newark Airport and areas of heavy industrialization and heavy traffic on the Turnpike. There are also areas of dense population on the east side of Newark Bay in the cities of Bayonne and Jersey City. The 1990 population within 16 km (10 mi) of the port terminals was 3,223,038. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 316,000; Oak Ridge Reservation, 290,000; Idaho National Engineering Laboratory, 536,000; Hanford Site, 585,000; and Nevada Test Site, 782,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,320 km (828 mi); Oak Ridge Reservation, 1,190 km (738 mi); Idaho National Engineering Laboratory, 3,860 km (2,396 mi); Hanford Site, 4,520 km (2,812 mi); and Nevada Test Site, 4,300 km (2,672 mi). Distances along rail routes are slightly longer.

There are no known special environmental concerns in the greater New York/New Jersey port area. The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Elizabeth, the Uniform Building Code requires buildings to withstand wind speeds up to 140 km/hr (85 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

Climatic Conditions

New York Harbor is located on the Atlantic coastal plain at the mouth of the Hudson River. The terrain is flat and diversified by numerous waterways; all but one of the city's five boroughs are situated on islands. Elevations range from less than 15 m (50 ft) over most of Manhattan, Brooklyn, and Queens to almost 90 m (300 ft) in the northern part of Manhattan and the Bronx, and over 120 m (400 ft) in Richmond (Staten Island).

Despite its nearness to the ocean and the numerous bays and rivers nearby, the port has a climate which more closely resembles the continental type of climate than it does the maritime type. Its modified continental climate follows from the fact that weather conditions affecting the city usually approach from a westerly direction and not from the ocean on the east. Some important exceptions to this must be noted, since the oceanic influence is by no means entirely absent. During the summer, local "sea breezes," winds blowing onshore from the cool water surface often moderate the afternoon heat; and most often in winter, coastal storms, accompanied by easterly winds, produce, on occasion, considerable amounts of precipitation.

From November through April the prevailing winds are from the northwest; for the remainder of the year the prevailing winds are southwesterly. Gales with velocities of 64 km/hr (40 mph) or more are predominantly from the northwest.

The mean annual temperature is slightly higher than that of most places in the United States of the same latitude, with the exception of localities near the Pacific coast. Precipitation is both moderate and distributed evenly throughout the year. Most of the rainfall from June through September comes from thunderstorms, therefore, is usually of brief duration, but relatively intense. From October to April, however, precipitation is generally associated with widespread storm areas, so that day-long rain or snow is common. Over the entire year, the city receives 59 percent of the sunshine hours possible at its latitude. This value compares favorably with that for any region east of the Mississippi, except the Southeast. Relative humidity averages about 66 percent for the year, showing that the city has a relatively damp climate.

Winds play an important role by affecting currents in the harbor. During the winter, west and northwest winds prevail, with northerlies and southwesterlies in secondary roles. The strongest winds are out of the west through northwest at 13 to 15 knots, from January through April. The sheltering effect of the land is apparent when looking at frequencies of winds of 28 knots or more. These winds blow at Ambrose Light about eight to nine percent of the time compared to one percent at Kennedy Airport and Floyd Bennet Field. Summer winds are often out of the south and southwest with a 10 to 12 knot afternoon peak. Fog in the harbor area is more closely related to land-type fogs. In winter, fog is common on clear, calm mornings and more frequent than at Ambrose Light. Southerlies can also bring winter fogs of the advection type. During the spring and early summer, the harbor and its approaches are susceptible to advection fog, riding in on east through south winds. A morning peak still exists in the harbor, while Ambrose Light exhibits an afternoon maximum (DOC, 1993b).

D.2.2.5 Fernandina Beach, FL

The Port of Fernandina is located about 9.3 km (5 mi) above the Entrance Seabuoy to the St. Marys River and Cumberland Sound. The entrance is bordered by two jetties on the approach to the cities of Fernandina Beach (located on Amelia Island) and St. Marys, GA, the Naval Submarine Base in Kings Bay, and an inland passage to St. Andrew sound via the Cumberland River (DOC, 1993d). The entrance is approximately 37 km (20 mi) north of the entrance to the Port of Jacksonville, which is located on the St. Johns River. A map of the port is shown in Figure D-36. Amelia Island is a small, historic, coastal resort town. Fort Clinch, a State Park, museum, and recreation area is located on the north end of Amelia Island at the inshore end of the south entrance jetty (DOC, 1993d).

The Port of Fernandina is a forest products and general cargo container port. It handles around 25,000 20-ft equivalent units of containerized freight and about 272,000 metric tons (299,000 tons) of forest products annually, but the container volume has varied considerably from year to year. Much of the port's trade is with South and Central America. There is also eastbound monthly service to the Mediterranean (Southern Shipper, 1993; American Shipper, 1994; Stubbs, 1994). Reportedly, the current controlling depth of the entrance, and that of approach channel to the submarine base, is 14.3 m (47 ft) and the controlling width is 122 m (400 ft). The same width channel with 10.3 m (34 ft) controlling depth is available to the Ocean Highway and Port Authority Terminal in Fernandina Beach. There is a 1.8 m (6 ft)

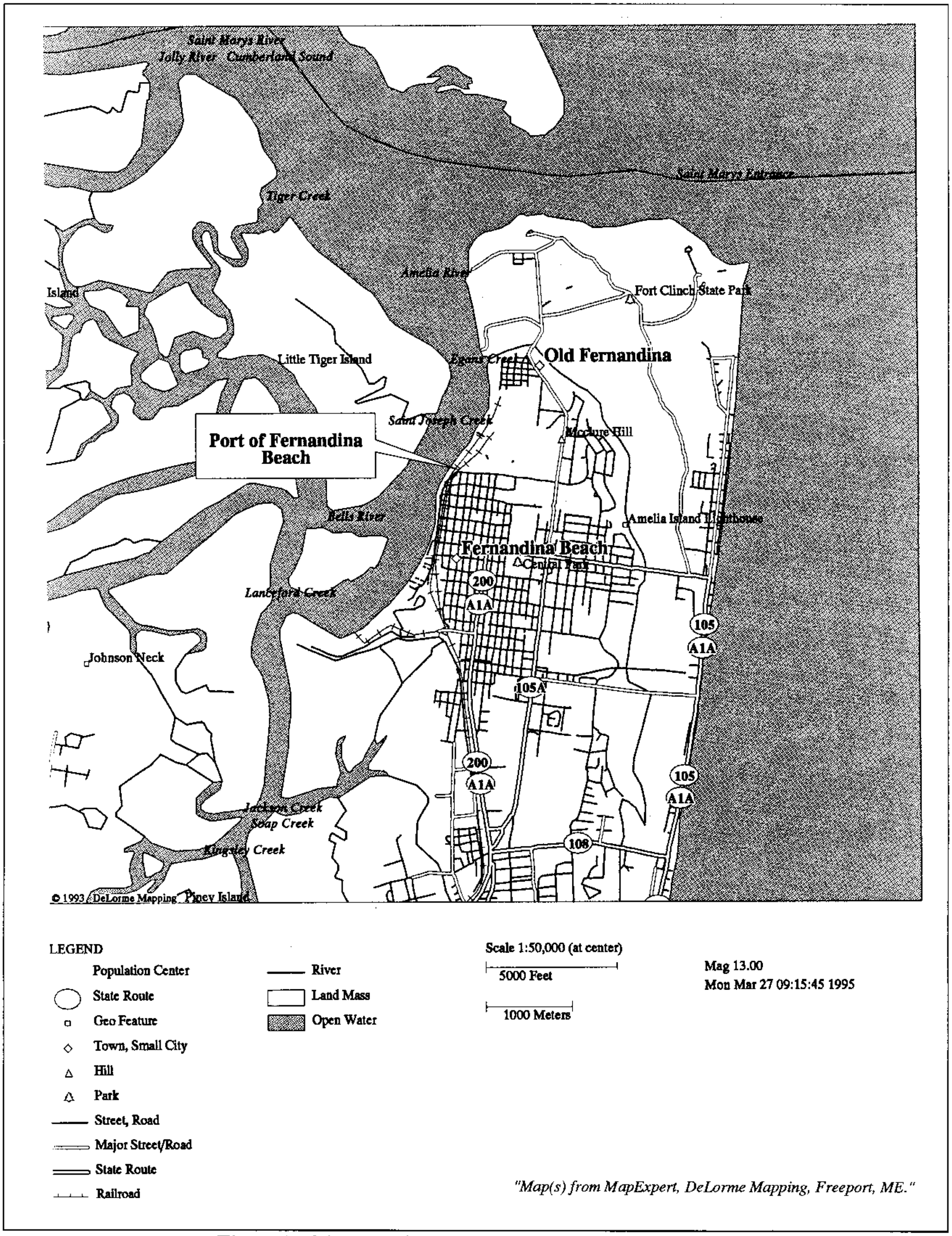


Figure D-36 Map of the Port of Fernandina Beach, FL

range of tide at Fernandina Beach. Tidal currents at the entrance have considerable velocity and are dangerous at times. A strong current "set" occurs at the St. Mary's entrance under certain weather conditions, it has been reported as high as 2.6 mi-per-sec (5 knots) (DOC, 1993d).

The Forest Products Terminal, located about 12 km (8 mi) above the channel entrance, is a publicly owned facility operated by Nassau Terminals, a private terminal operating and stevedoring company. Following a port expansion in 1992, the Terminal consists of 366 m (1,200 ft) of useable berthing situated on the left ascending bank of the Amelia River. The new capacity of the port is about 50,000 20-ft equivalent units per year. The Terminal is equipped with two 36 metric ton (40 ton) container cranes and other container handling equipment, a 4,645 m^2 (50,000 ft^2) container freight station, 2.0 ha (5 acres) of open storage area, and is served by the CSX Railroad with pierside rail trackage (DOC, 1993d; Southern Shipper, 1993). The port handles an average of two vessels a day, typically a cruise vessel and a cargo vessel. The only products normally handled by the port are forest products for a paper mill located in the area, and containers loaded with food and paper products. The passenger or cruise ship business is small, using smaller vessels for cruises in the near islands and offshore (Robas, 1994).

The port terminal is located in the downtown section of the town of Fernandina Beach. Truck access to the port is through the downtown area and mixed residential/business structures for a distance of about 8 km (5 mi). Total distance to Interstate 95 is about 24 km (15 mi), much of which is divided multi-lane highway of mostly rural character.

Other Pertinent Port Information: Terminal property is fenced and lighted and has 24-hour watchman service. Rail openings into the port are not secured. The port has little experience in handling hazardous materials, in that hazardous materials are not normally shipped in or out of the port (Robas, 1994).

The U.S. Army Corps of Engineers was to award a contract in October 1994 for deepening the harbor channel to 11 m (36 ft) and constructing a 366 m (1,200 ft) turning basin. The approach channel to the Terminal passes through a State aquatic preserve for the manatee and other marine animals. Nassau Terminals occasionally handles some containerized hazardous materials; however, a port official thought there would be considerable local opposition to handling spent nuclear fuel shipments for fear of the effect of adverse publicity on tourism in this popular resort area (Stubbs, 1994).

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Fernandina Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). The port is located in a low seismic zone with an acceleration of 0.075 g or less.

The 1990 population within 16 km (10 mi) of the port terminals was 32,952. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 45,000; Oak Ridge Reservation, 185,000; Idaho National Engineering Laboratory, 590,000; Hanford Site, 650,000; and Nevada Test Site, 650,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 610 km (380 mi); Oak Ridge Reservation, 920 km (570 mi); Idaho National Engineering Laboratory, 4,000 km (2,500 mi); Hanford Site, 4,700 km (2,900 mi); and Nevada Test Site, 4,200 km (2,600 mi). Distances along rail routes are slightly longer for Western sites, and about the same for Eastern sites.

Environmental Conditions

The State of Florida has classified the Amelia River near the St. Mary's Entrance as a Class III water body. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). In addition, the State of Florida has designated certain waters in the vicinity of Fernandina Beach as "Outstanding Florida Waterways", which are afforded special protection. Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Waters located near the Port of Fernandina Beach designated as Outstanding Florida Waters include Fort Clinch State Park, Fort Clinch State Park Aquatic Preserve, Nassau Valley State Reserve, and the Nassau River-St. Johns River Marshes Aquatic Preserve (FL DEP, 1994).

The Amelia River, in the vicinity of the Port of Fernandina Beach, is characterized as a mid- salinity estuarine habitat (generally 5 to 16.5 parts per thousand). There are both commercial and recreational fish and invertebrates found in the vicinity of the port. These aquatic species include: blue crabs, shrimp, American eel, Atlantic menhaden, tarpon, sea catfish, sheepshead, spotted seatrout, weakfish, spot, Atlantic croaker, kingfish, drum, flounder, silver perch, bluefish, mullet, pinfish, pigfish, ladyfish, and snapper (FWS, 1980e).

The Fort Clinch State Park and Fort Clinch State Park Aquatic Preserve are located on Amelia Island adjacent to Fernandina Beach. Birds that can be found in Fort Clinch State Park include various types of shorebirds, wading birds, waterfowl, raptors, songbirds, and seabirds. Endangered or threatened bird species in Fort Clinch State Park include: brown and white pelican, great egret, snowy egret, tricolored heron, little blue heron, black-crowned night heron, yellow-crowned night heron, least bittern, wood stork, white ibis, bald eagle, northern harrier, osprey, American kestrel, merlin, peregrine falcon, clapper rail, piping plover, American oystercatcher, least tern, black skimmer, royal tern, caspian tern, sandwich tern, worm-eating warbler, yellow-throated warbler, prairie warbler, Louisiana waterthrush, and American redstart (Fort Clinch State Park, 1994). Species with special status found in the area include the loggerhead sea turtle, the manatee, the American alligator, the least tern, and the burrowing four-o'clock (Murray, 1994). The loggerhead sea turtle, a Federally protected species, uses much of Amelia Island and Cumberland Island as nesting areas. In addition, the U.S. Fish and Wildlife Service reports that the following protected marine species may occur in Nassau County: west indian manatee (endangered), Kemp's ridley sea turtle (endangered), leatherback sea turtle (endangered), loggerhead sea turtle (threatened), hawksbill sea turtle (endangered), and the green sea turtle (threatened). Protected bird species include the wood stork (endangered) and red-cockaded woodpecker (endangered) (Bentzien, 1994).

Climatic Conditions

As with the other more northern ports, the climate of this area is also modified by the influence of the Atlantic Ocean. Easterly winds occur roughly 40 percent of the time, producing a true maritime climate for the area. The greatest rainfall occurs during summer, usually associated with afternoon and evening thunderstorms. During summer, measurable precipitation can be recorded nearly every two days. The prevailing winds are northeasterly in the fall and winter months and become more southwesterly during spring and summer. Although this region is located along the eastern Florida coast, it has been very fortunate in escaping hurricane-force winds. The majority of systems in recent years that have reached this latitude have moved parallel to the coastline, keeping well offshore. Others have weakened significantly moving over land prior to reaching the area. The combination of these two factors has spared the area from any major devastation due to tropical systems in recent years (NOAA, 1992e).

D.2.2.6 Freeport, TX

Freeport harbor is located about 64 km (40 mi) southwest of the Galveston, Texas harbor entrance, and about 5 km (3 mi) from the Gulf of Mexico on the Brazos River (DOC, 1992a), with the Gulf Intracoastal Canal crossing the river, making deepwater activity available. The main channel is maintained at 13.6 m (45 ft) and leads to a 364 m (1,200 ft) turning basin (D&B, 1993). Freeport is principally involved in petroleum and petrochemical transport (AAPA, 1994). However, in 1992, 188,400 metric tons (207,711 tons) of containerized cargo (approximately 20,000 20-ft equivalent units) were handled in the port. Primary inbound cargoes were bananas and fruit, and primary outbound cargoes were rice and chemicals (AAPA, 1994).

The harbor is regulated by the Navigation and Canal Commissioners of the Brazos River Harbor Navigation District, and is known locally as Brazosport (DOC, 1992a). The ship channel has been improved by construction of jetties on either side of the entrance. A map of the port is shown in Figure D-37.

Berth assignments at the Port of Freeport are made by the Terminal Superintendent. The port has five general breakbulk berths, 664 m (2,190 ft) in length with 10.9 m (36 ft) depth alongside. There is 19 ha (47 acres) of open storage adjacent to the wharves (D&B, 1993). The port has rail facilities with dual tracks on Berths 1, 1A (Brazos Harbor Public Facility Wharf), and 2 (Brazos River Harbor Wharf No. 2). Both facilities have substantial covered storage available for short-term storage. General cargo is usually handled by the ship's tackle, and no container cranes are available at the port [a floating 450 metric ton (500-ton) derrick is available for heavy lifts by special arrangement] (DOC, 1992a; AAPA, 1994).

Highway connection from the port is via State Highways 227 and 288, for approximately 56 km (35 mi) to Houston, where Interstate-10 is accessed.

Other Pertinent Information: There are no known restrictions on receipt of foreign research reactor spent nuclear fuel at the port, but there are substantial conflicting activities at the port, including petrochemicals and hazardous chemicals (AAPA, 1994). The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Freeport, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census population of Freeport was 12,600. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 360,000; Oak Ridge Reservation, 300,000; Idaho National Engineering Laboratory, 480,000; Hanford Site, 530,000; and Nevada Test Site, 530,000. Populations along rail routes to these sites are slightly higher for Savannah River Site and Oak Ridge Reservation, but slightly lower for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site. The approximate distances to the five potential sites on interstate routes are: Savannah River Site, 1,600 km (1,000 mi); Oak Ridge Reservation, 1,600 km (1,000 mi); Idaho National Engineering Laboratory, 3,100 km (1,900 mi); Hanford Site, 3,700 km (2,300 mi); and Nevada Test Site, 3,100 km (1,900 mi). Distances along rail routes are about the same.

Climatic Conditions

Weather in this area is only an occasional navigational problem. Winds blow at 28 knots (32 mph) or more approximately 3 to 4 percent of the time in November and from January through April. Average speeds are 12 to 14 knots (14 to 16 mph) during this period. Fog is also a winter problem, and visibilities drop below 160 m (0.25 mi) on approximately three to six days each month from November through April.

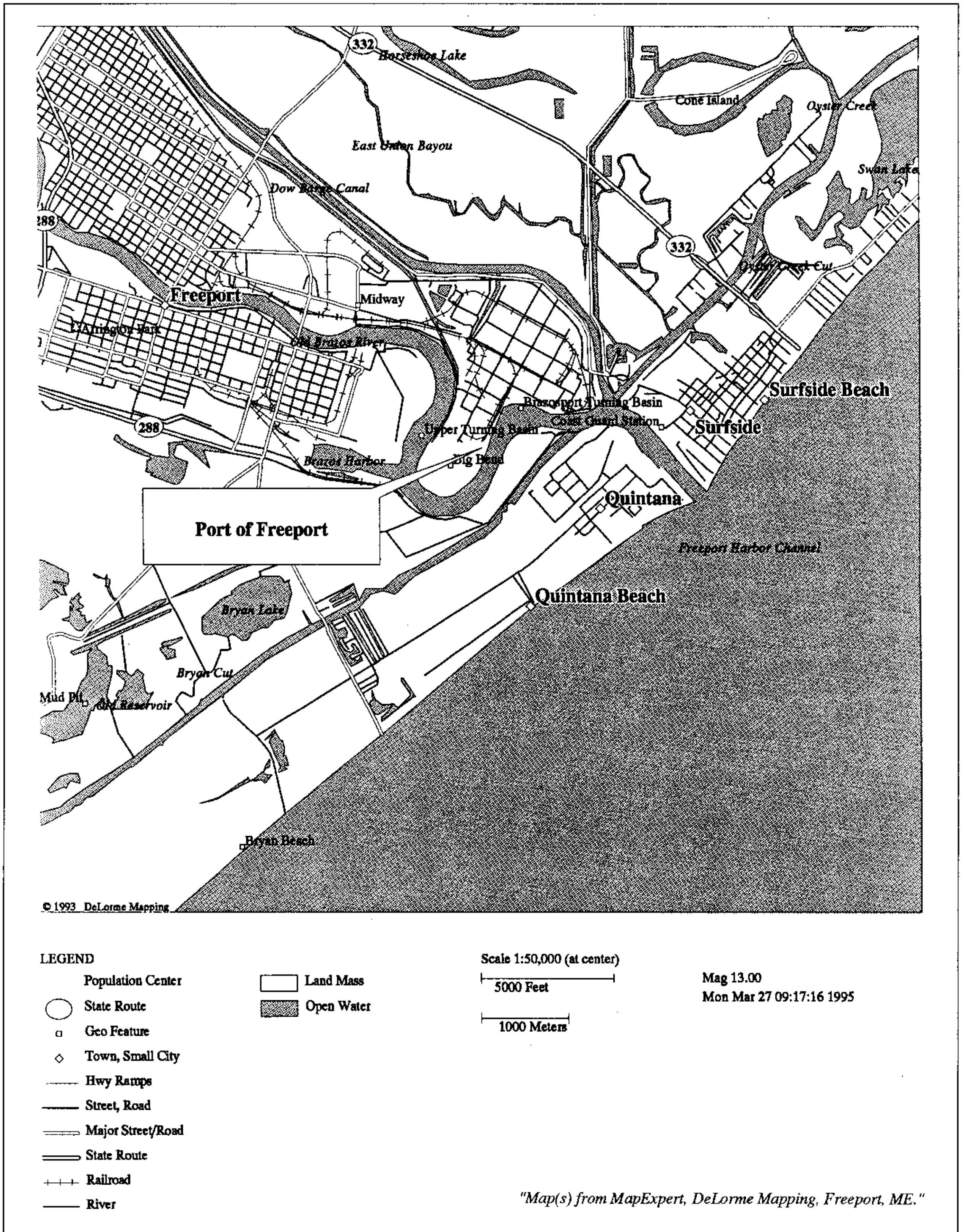


Figure D-37 Map of the Port of Freeport, TX

Thunderstorms are most frequent from April through September, during the afternoon and evening. These thunderstorms are usually air mass types as opposed to the less frequent but more severe thunderstorms that occur with fronts and squall lines from fall through spring. Tropical cyclones, particularly severe hurricanes, are most likely in August and September (DOC, 1992a).

D.2.2.7 Gulfport, MS

Gulfport, the seat of Harrison County, MS, is a seaport and tourist center located on the north side of Mississippi Sound, approximately 26 km (16 mi) from the entrance to the Ship Island Bar Channel on the Gulf of Mexico. Gulfport is located approximately 97 km (60 mi) east of New Orleans, LA. The approach to Gulfport is through a dredged channel marked by lighted buoys. Federal project depths are 9.7 m (32 ft) for the bar channel and 9.1 m (30 ft) for the Gulfport Channel and Harbor Basin. The harbor was deepened to 10.97 m (36 ft) mean low water in 1993 (DOC, 1992a; AAPA, 1993; Southern Shipper, 1993). A map of the port is shown in Figure D-38.

The State-owned Port of Gulfport is a small, but growing, niche port on the Gulf Coast primarily handling containerized banana imports and dry bulk commodities. The port has a growing general cargo outbound container tonnage as the fruit carriers fill otherwise empty containers on the return leg of voyages. Container traffic for fiscal year 1991 included 68,000 20-ft equivalent units amounting to approximately 664,973 mt (733,000 ton) of cargo (AAPA, 1993). By 1993, container volume increased to 736,100 mt (811,559 tons), or approximately 75,000 20-ft equivalent units (AAPA, 1994).

Gulfport has 10 berths with a total of 1,768 m (5,800 ft) of lineal berthing space. There is an open storage area of four ha (10 acres) and a shed area of 19,000 m^2 (204,500 ft^2). A second container berth (East Pier) is used for self-contained container ships. The port's West Pier container berth is approximately 750 m (2,460 ft) long, whereas the East Pier is approximately 200 m (656 ft) long. Gulfport has two 30.5 metric ton (34 ton) container cranes at its West Pier (AAPA, 1994).

The port is located immediately adjacent to the City of Gulfport, which forms the northern boundary of the terminal area. The Terminal has almost immediate access to U.S. Highway 90, and is about 5 mi from I-10, a major east-west roadway. U.S. Highway 49, which begins at the terminal gate and connects with I-10, runs through the center of the City. The port is served by the CSX and Mid-South Railroads with connections to the Norfolk Southern at Hattiesburg. Double trackage extends to the container berth.

Gulfport is presently served by one common carrier combination container/breakbulk ship operator, ABC Line, which operates five large ships on North European around-the-world trade routes. Three other containerized fruit carriers also regularly call at Gulfport (AAPA, 1994; Southern Shippers, 1993).

Other Pertinent Information: Gulfport employs a port security firm that maintains 24-hour guard service. The port is fenced with controlled access to vehicles and personnel. It does not appear that there are any regulations preventing the importation of spent nuclear fuel, although the port indicates that the Coast Guard may impose bans on especially hazardous shipments (Edwards and Burns, 1993). Gulfport has no prior experience handling spent nuclear fuel (NRC, 1993; SNL, 1994) and, as far as is known, there are no hazardous cargoes routinely handled at Gulfport. Port personnel provide First Response augmented by the Gulfport Fire and Police Departments. The Port of Gulfport conducts hazardous materials training of port personnel (Edwards and Burns, 1993). There is a former small cruise ship terminal at the East Pier as well as a floating casino located near the street entrance to the port. The United States Coast Guard indicated that the East Dock is presently slated for casino development, and there are two casinos on the north end of

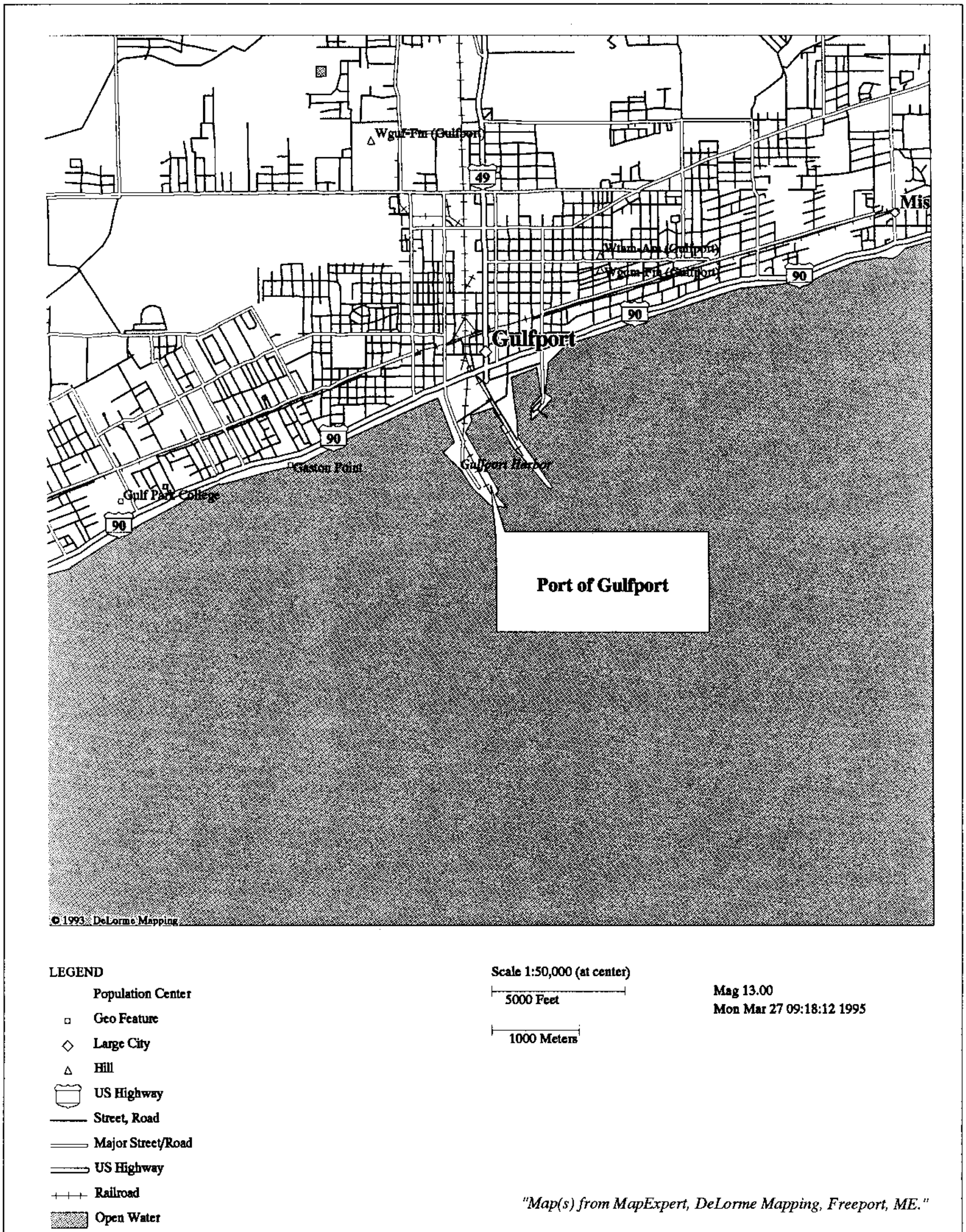


Figure D-38 Map of the Port of Gulfport, MS

the West Dock. As a result, the hazardous materials area at the north end of the West Dock has been eliminated for explosives. Also, the facility of particular hazard cannot be used for foreign research reactor spent nuclear fuel, and is not secure or well lit. (Brown, 1995)

There are no known sanctuaries or wildlife habitats in the immediate port area. However, to enter Gulfport, ships must pass close to the protected Gulf Islands National Seashore. The port is subject to severe hurricane and tropical storms. The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Gulfport, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of 0.075 g or less.

The 1990 population within 16 km (10 mi) of the port terminals was 113,153. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 194,000; Oak Ridge Reservation, 146,000; Idaho National Engineering Laboratory, 435,000; Hanford Site, 484,000; and Nevada Test Site, 683,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 910 km (565 mi); Oak Ridge Reservation, 920 km (573 mi); Idaho National Engineering Laboratory, 3,570 km (2,219 mi); Hanford Site, 4,240 km (2,635 mi); and Nevada Test Site, 3,530 km (2,195 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Mississippi Sound is separated from the Gulf of Mexico by a series of uninhabited barrier islands. Vessels approaching the port from the Gulf enter the Ship Island Channel, which runs between the west end of Ship Island and Ceat Island and the east end of Cat Island. Ship Island, along with Horn Island and Petit Bois Island, which are further to the east, comprise the Gulf Islands National Seashore. These islands serve as a wintering and migratory area for the protected Peregrine Falcon as well as various shorebirds. Ships entering the Ship Island Channel from the Gulf pass directly north of the northern end of the Chandeleur Islands that comprise the Breton National Wildlife Refuge and Breton Wilderness. The Breton National Wildlife Refuge and Breton Wilderness, which includes areas of black mangroves, serves as a breeding area for the protected loggerhead sea turtle and brown pelican, and a migratory area for the protected peregrine falcon. This area is also home to a variety of shorebirds, wading birds, waterfowl, raptors, seabirds, and songbirds (FWS, 1982a).

The U.S. Fish and Wildlife Service reported that several Federally-listed protected species may occur in the Port of Gulfport area. These species include the endangered brown pelican and Kemp's ridley sea turtle and the threatened gulf sturgeon and loggerhead sea turtle (Goldman, 1994). According to the Mississippi Natural Heritage Program, four protected species of bird have been spotted feeding or loafing in the area of the Port of Gulfport. These species include the royal tern, black rail, reddish egret, and the piping plover (Gordon, 1994). Commercial harvesting areas for the eastern oyster are located throughout the Mississippi Sound, including several areas within a few miles of the port. Breeding areas for the State-protected Least tern also are located along the coast of Gulfport (FWS, 1982a).

The waters in the vicinity of the port have been classified by the State of Mississippi for "recreation" but not as a water supply (Reaves, 1994). The nearshore waters of the Mississippi Sound are characteristic of a middle salinity estuarine habitat (generally 5 to 20 parts per thousand). Aquatic organisms that are typically found in the waters of this area include: shrimp, blue crab, seatrout, croaker, drum, spot, kingfish, flounder, catfish, mullet, Florida pompano, bluefish, Gulf menhaden, bay anchovy, Crevalle jack, blue runner, Alabama shad, and Atlantic bottlenose dolphin (FWS, 1982a).

Climatic Conditions

Because of Gulfport's geographic location, the local weather is greatly influenced by the Gulf of Mexico. Generally, summers are warm but temperatures are more moderate than those observed at inland locations because of the diurnal sea breeze circulation. Winter weather is generally mild, with the exception of the occasional cold air outbreak. These events occur at 3-10 day intervals between October and April in the Gulf of Mexico region, generally lasting less than three days. The annual rainfall in this region is among the highest in the continental United States. The precipitation is fairly evenly distributed throughout the year with a maximum coinciding with the summer thunderstorm season and minimum occurring during the late Fall months. However, extended rainy periods are rare in this region. Thunderstorm frequencies are highest in July and August, where they may occur every other day, but rarely do they reach intense or violent levels. However, the area is quite vulnerable to tropical systems (e.g., Hurricane Camille, 1969), which originate in the West Indies, West Caribbean, and the Gulf of Mexico (NOAA, 1992i; Wayland and Raman, 1989).

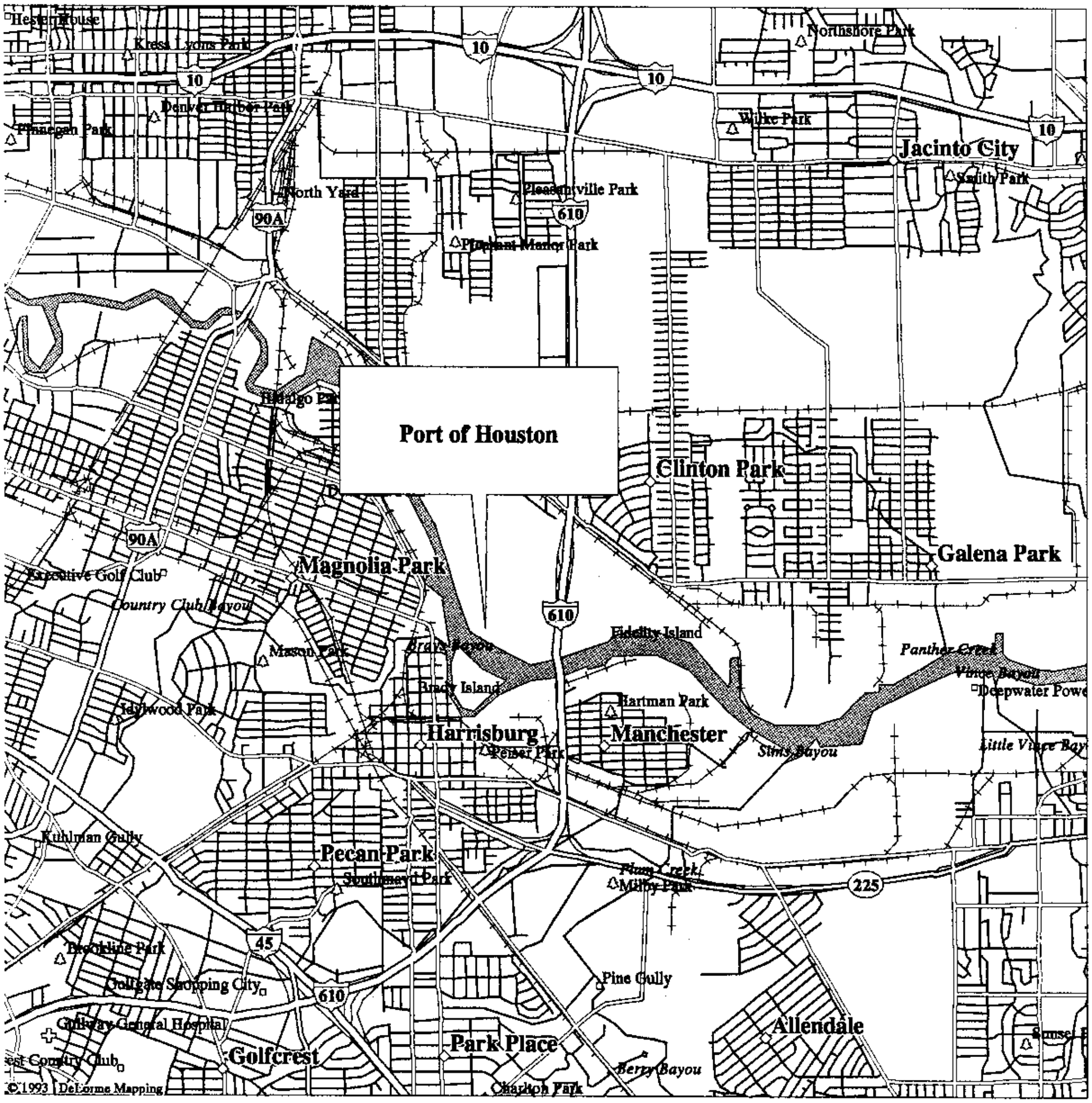
D.2.2.8 Houston, TX

Houston is the largest city in the State of Texas, and the Port of Houston is one of the largest ports in the United States (in terms of total tonnage handled). Morgans Point, approximately 37 km (23 mi) from the entrance to Galveston Bay, marks the beginning of an extensive industrial area lining the Houston Ship Channel. Houston is at the head of the channel, 71 km (44 mi) from the Gulf of Mexico. The transit of large ships is restricted to the ship channel across Galveston Bay and through parts of the San Jacinto and Buffalo Bayou. A Federal project provides for a 12.2 m (40 ft) channel from the entrance to the Gulf of Mexico to Houston (Brady Island) DOC, 1992a). The Houston Ship Channel is 12.2 m (40 ft) deep, and its width ranges from about 76 m to 120 m (250 to 400 ft), making the transit difficult for the large number of ship transits each year (D&B, 1993). A map of the port is shown in Figure D-39.

The Port of Houston is a 40.2 km-(25 mi)-long complex of diversified public and private facilities located on both banks of the Houston Ship Canal, which empties into and transits Galveston Bay. Bulk cargoes, dry and liquid, (including petroleum and petrochemicals) comprise the major share of tonnage handled by the port. Estimated tonnage for 1992 amounted to a total of 114.3 million metric tons (126 million short tons), of which bulk accounted for 72 percent, breakbulk 6.6 percent, and container 3.4 percent.

The Houston Port Authority owns and operates six public cargo facilities including: the Turning Basin Terminal (general cargo) located at the head of the Houston Ship Channel; Jacintoport Terminal (general cargo) located on the north side of the channel near Channelview, Texas; and Barbours Cut Container Terminal located at the head of Galveston Bay on the left ascending bank of the Houston Ship Channel (Jane's, 1992; AAPA, 1993; POHA, 1993).

Barbours Cut Container Terminal: About 40 km (25 mi) from the entrance to the ship channel, this terminal is designed to handle containers, roll-on/roll-off ships. The Barbours Cut Container Terminal is equipped with eight container cranes and five container berths [each is 305 m (1,000 ft) long], plus a separate roll-on/roll-off terminal. The terminal occupies 87 ha (215 acres) of developed land, including 17.8 ha (44 acres) of paved marshalling area for roll-on/roll-off cargoes. Barbours Cut Container Terminal



LEGEND

Population Center	River	
◇ Town, Small City	Land Mass	
Δ Park	Open Water	
Interstate, Turnpike		
Hwy Ramps		
Street, Road		
Major Street/Road		
Interstate Highway		
US Highway		
Railroad		

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:21:36 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-39 Map of the Port of Houston, TX

has depths of 12.2 m (40 ft) at mean low water. Crane capacity for Barbours Cut Container Terminal is six 40.6 metric ton (45 ton) container cranes and two 30.5 metric ton (34 ton) container cranes (Jane's, 1992; AAPA, 1993).

Turning Basin Terminal: This terminal has several berths, the largest of which is 243.8 m (800 ft) long and can handle a 228.6 m (750 ft) ship. Turning Basin Terminal's depths are 10.97 m (36 ft) at mean low water. Crane capacity for this terminal is one 40.6 metric ton (45 ton) container crane and one 76.2 metric ton stiff-leg crane.

Barbours Cut Container Terminal has three entry points (gates) with a total of 21 truck lanes that are reached via Barbours Cut Boulevard, a multi-lane limited access roadway. Access to I-610, the Houston Beltway, and other interstate highways is via State Highway 146, which connects with State Highway 225 about 4.8 km (3 mi) from the Terminal. The Route 225 connector is an east-west highway about 22.5 km (14 mi) long. It appears that these routes run through commercial/residential areas with the opportunity for congestion. Barbours Cut Container Terminal is served by the Port Terminal Railroad Association and the Santa Fe Railroad. The Railroad Association connects with all other railroads including the Southern Pacific, Union Pacific, Burlington Northern, and the Houston Belt and Terminal Railroad. Trailer-on-Flat-Car shipments are possible within the terminal, but trackage does not extend to the container berths (Jane's, 1992; AAPA, 1993).

Barbours Cut Container Terminal is host to a large number of major international container and roll-on/roll-off ship lines. A partial listing includes: ABC Container Line, A. Bottacchi, ACL/Gulf Container Line, Afram Lines Ltd, America/Africa/ Europe Line, Atlantic Cargo Services, Baltic Shipping Co., Bank Line, Barber Blue Sea, CGM, CNAN, Columbus Line, COSCO, Costa Container Service, DB Turkish Cargo Line, Djakarta Lloyd, East Asiatic Ltd, Ellerman Line, Farrell Lines, Gulf Mideast Lines, Hapag-Lloyd, Hoegh Lines, Hyundai Merchant Marine, Italian Line, Ivaran Lines, Jugolinija, Kingwood Container Line, Maersk Line, Mediterranean Shipping Company, Nedlloyd Lines, SafBank Line, Sea-Land, Shipping Corp of India, Delmas-Vieljeux, Spanish Line, Torm Lines, Trans Freight Lines, United Arab Shipping Co., Waterman-Isthmian Line, and Zim Line (Jane's, 1992, AAPA, 1993).

Other Pertinent Information: The Houston Port Authority has its own 24-hour security force and all of its terminals are fenced with controlled access. A fireboat is stationed at Barbours Cut Container Terminal, which also has a full-service fire department. There is space within Barbours Cut Container Terminal for temporary segregation of hazardous cargoes (Horan, 1993).

A Port Authority Official was unaware of any regulations prohibiting the importation of spent nuclear fuel. The Houston Port Authority handles a lot of hazardous cargoes including radioactive substances, but the official did not know if the port has ever handled spent nuclear fuel (Horan, 1993). Available data indicates the port has not handled spent nuclear fuel at least since 1979 (NRC, 1993; SNL, 1994). The Houston Ship Channel and Galveston Bay are host to many petroleum and petrochemical berths and terminals served by a large amount of tanker and tank barge traffic. Many of these facilities are located upstream of Barbours Cut Container Terminal, which does not appear to have any conflicting use within its boundaries. The Houston Port Authority has its own emergency response team and fire department. The Houston Fire Department's hazardous materials team is used as a backup in emergencies. The Houston Port Authority has a hazardous materials training program for its terminal operating personnel. It is not known if longshoremen also receive this training (Horan, 1993).

There have been a number of ship accidents, tanker fires, and pipeline accidents at facilities near the Port of Houston in recent years. The United States Coast Guard data indicates that for the period 1991 to 1993, there were about 7,100 ship transits of the channel that resulted in 32 collisions, 33 allisions, 5 ship fires, and 59 hard groundings (USCG, 1994b). Because the accident statistics also reflect barge traffic risks, the accident rate for oceangoing vessels is probably lower, but there is not data to refine that estimate available yet.

The Turning Basin Terminal is located at the terminus of the Ship Channel in a densely populated area above all other public and private terminal facilities within the port. Barbours Cut Container Terminal is remotely located from the City of Houston with relatively good separation from other terminals and traffic in the area (see II.E above). However, the two small communities of Morgan Point and La Porte (with a population of about 20,000) are located adjacent to the Terminal on the south.

There are no special plant, fish, or wildlife sanctuaries in the vicinity of Barbours Cut Container Terminal. The port is subject to hurricanes and tropical storms. The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Houston, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 census city population was 1,630,553, with the density estimated at 1,083 persons/km^2. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 349,000; Oak Ridge Reservation, 283,000; Idaho National Engineering Laboratory, 471,000; Hanford Site, 579,000; and Nevada Test Site, 524,000. Populations along rail routes to these sites are slightly smaller for Idaho National Engineering Laboratory, Hanford Site and Nevada Test Site, but slightly larger for Savannah River Site and Oak Ridge Reservation. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,550 km (964 mi); Oak Ridge Reservation, 1,480 km (918 mi); Idaho National Engineering Laboratory, 3,000 km (1,866 mi); Hanford Site, 3,610 km (2,282 mi); and Nevada Test Site, 2,930 km (1,818 mi). Distances along rail routes are about the same except for Savannah River Site, which is slightly longer.

Climatic Conditions

The climate of Houston is predominantly marine. The terrain includes numerous small streams and bayous, which together with the nearness to Galveston Bay, favor the development of both ground and advective fogs. Prevailing winds are from the southeast and south, except in January, when frequent passages of high-pressure areas brings invasions of polar air on prevailing north winds.

Temperatures are moderated by the influence of winds from the Gulf, which results in mild winters and, on the whole, relatively cool summer nights. Another effect of the nearness of the Gulf is abundant rainfall, except for rare extended dry periods. Polar air penetrates the area frequently enough to provide stimulating variability in the weather.

The average number of days with minimum temperatures of 32°F or lower is only about 7 per year at the city's National Weather Service office and about 15 per year at William P. Hobby Airport, which is about 16 km (10 mi) southeast of the city. Most freezing temperatures last only a few hours because they are usually accompanied by clear skies.

Monthly rainfall is evenly distributed throughout the year. In past years, about 75 percent of the total precipitation has been between 76.2 and 152.4 cm (30-60 in). Since thundershowers are the main source of rainfall, precipitation may vary substantially in different sections of the city on a day-to-day basis.

Records of sky cover for daylight hours indicate about one-fourth of the days per year as clear with maximum of clear days in October. Cloudy days are relatively frequent from November to May, and partly cloudy days are more frequent from June through September.

Snow rarely occurs; however, on February 14-15, 1895, 51 cm (20 in) of unmelted snow was measured. Heavy fog occurs on an average of 16 days a year, and light fog occurs about 62 days a year in the city, but the frequency of heavy fog is considerably higher at William P. Hobby Airport. Destructive windstorms are fairly infrequent, but both thundersqualls and tropical storms occasionally pass through the area (DOC, 1992d).

D.2.2.9 Lake Charles, LA

The city of Lake Charles, the seat of Calcasieu Parish, is located on the east side of the Lake. It is the center of large industries such as chemical, petroleum, natural gas, fish oil, synthetic rubber, salt, seafood, and rice. The Port of Lake Charles is situated 3 km (2 mi) south of the city on the east bank of the Calcasieu Lake, and is 52 km (32 mi) from the Gulf of Mexico (DOC, 1992a). A map of the port is shown in Figure D-40.

A Federal project provides for a channel 12.8 m (42 ft) deep across the outer bar, from 12.2 to 12.8 m (40 to 42 ft) through the jetties, and 12.2 m (40 ft) to the Port of Lake Charles.

The United States Coast Pilot (DOC, 1992a) reports: "In recent years a substantial number of oceangoing vessels of increased size and draft have been entering the Calcasieu River Channel and proceeding to and from berths as far up the channel as the Port of Lake Charles. The channel, however, has not been appreciably widened in recent years. Based upon reported marine casualties to vessels and upon reported navigational problems arising from the increased oceangoing traffic, and after consultation with local marine interests, the Coast Guard Captain of the Port (COTP) has developed certain guidelines to enhance safe navigation."

The longest berth in the terminal is 274 m (900 ft). Lake Charles has no international container carriers serving the port and serves primarily as a breakbulk, dry bulk, and project cargo niche port (AAPA, 1993 and 1994; Southern Shipper, 1993). It can handle limited container traffic on breakbulk vessels (about 30,000 20-ft equivalent units in 1992) (Southern Shipper, 1993). Most of the area around Calcasieu Lake is wetlands, and ships entering the port pass by the Sabine National Wildlife Refuge.

Like all Gulf Coast ports, it is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Lake Charles, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Lake Charles, LA's climatic and environmental conditions are similar to those of the Port of New Orleans, LA. Port of New Orleans information is presented in Section D.2.2.14.

The 1990 census population estimate for this port vicinity was approximately 73,800 with a population density on the order of 940 persons/km^2 (2,434 persons/mi^2). The approximate distances to the five potential sites on interstate routes are: Savannah River Site, 1,100 km (700 mi); Oak Ridge Reservation, 960 km (600 mi); Idaho National Engineering Laboratory, 3,400 km (2,100 mi); Hanford Site, 4,000 km (2,500 mi); and Nevada Test Site, 3,200 km (2,000 mi). Distances along rail routes are about the same.

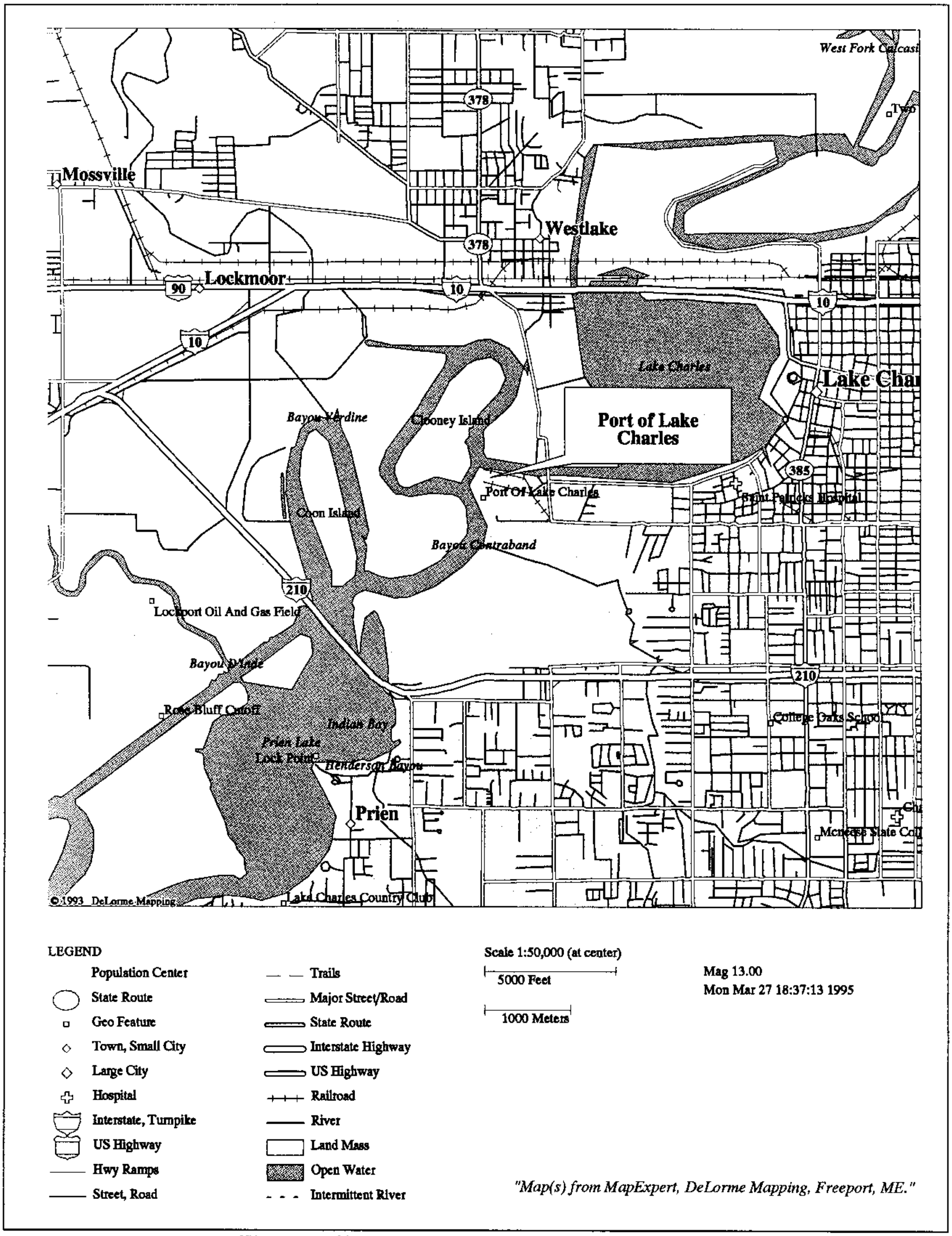


Figure D-40 Map of the Port of Lake Charles, LA

D.2.2.10 Long Beach, CA

Long Beach and Los Angeles Harbors, although divided by a political boundary, form a single geographic and economic port entity. The harbors occupy a major part of San Pedro Bay. The Port of Long Beach, one of the largest ports on the Pacific Coast, has extensive foreign and domestic traffic with modern facilities for the largest vessels. Most of the channels in Long Beach Harbor are maintained at more than the project depth of 10.7 m (35 ft). The entrance to Middle Harbor is 3.5 km (2.2 mi) from the Queens Gate entrance at the Pacific Ocean. The channel from the Pacific Ocean is straight, short, and direct (DOC, 1992b). A map of the port is shown in Figure D-41.

The Long Beach Harbor Department is a semi-autonomous agency of the City of Long Beach, CA. The Department is responsible for the operation, control, and development of the municipally owned port facilities. Long Beach is a large port (a load center) with 1,040 ha (2,816 acres) of land area, 12 piers, and 77 operational berths serving about 5,700 vessels annually. The port handles about 75 million metric tons (83 million tons) of revenue cargo annually, of which approximately 35 million metric tons (39 million tons) is containerized general cargo equivalent to 1.8 million 20-ft equivalent units (POLB, 1993a-d; AAPA, 1993).

Long Beach is a multi-terminal port and is host to seven container terminals with 38 container cranes and 243 ha (600 acres) devoted to container handling facilities. Additionally, there are facilities for petroleum and petroleum-related products, dry bulk materials, automobiles, steel, citrus, palletized general cargoes, and other commodities. The port functions as a "landlord" port leasing out its facilities to terminal and ship operating companies. Two of the container terminals (California United Terminals - Pier E, and Pacific Container Terminal - Pier J) are operated as "public" facilities. California United Terminals also has two roll-on/roll-off ramps and rail spurs (POLB, 1993a-d; AAPA, 1993; Janes's, 1992).

The dock/quay length available for cargo ships is as follows: California United, Pier E, Berths E24-E26 — 594 m (1,950 ft) long, and Pacific, Pier J, Berths J245-J247 — 1,006 m (3,300 ft) long. The corresponding depths alongside at mean low water are: California United with 14-15.2 m (46 to 50 ft), and Pacific Container with 14.9 m (49 ft). The five cranes at California United are all 40 metric ton (44 ton) container cranes. Pacific Container has six, 40 metric ton container cranes (Jane's 1992; AAPA, 1993; POLB, 1993a-d).

California United Terminals is served by an 11 lane main gate, which appears to be about 0.8 km (0.5 mi) from the "on" ramp to I-710 (the Long Beach Freeway), all within the confines of the port area. Pacific Container Terminal has similar ease of access to I-710, estimated to be a distance of about 1.9 km (1.2 mi), also within the port terminal. California United Terminals has shipside rail service provided by the Harbor Belt Line Railroad. Plans are to extend Belt Line rail service to Pacific Container Terminal by April 1994. The line connects with the major rail systems serving the Greater Los Angeles/Long Beach areas such as the Union Pacific, Santa Fe, and Southern Pacific Railroads. The latter operates a 97 ha (240 acre) intermodal container transfer facility which was built by the POLB to serve the marine terminals of both Long Beach and Los Angeles. The terminals are about 6.4 km (4.0 mi) from the double stack intermodal container transfer facility yard. The Santa Fe and Union Pacific railroads offer similar intermodal transfer facilities at their respective yards in east Los Angeles (D&B, 1993; Jane's, 1992; AAPA, 1993).

The port is served by a number of the world's largest container ship lines including: ACL, BHP/MTL, CCNI, COSCO, CGM, Cho Yang, Cool Carriers, DSR-Senator Line, EAC, Hanjin, Hapag-Lloyd, Hyundai, K-Line, Maersk Line, Nedlloyd Lines, OOCL, Philippine National Line, P & O, Sea-Land and TMM (Jane's, 1992; D&B, 1993; AAPA, 1993).

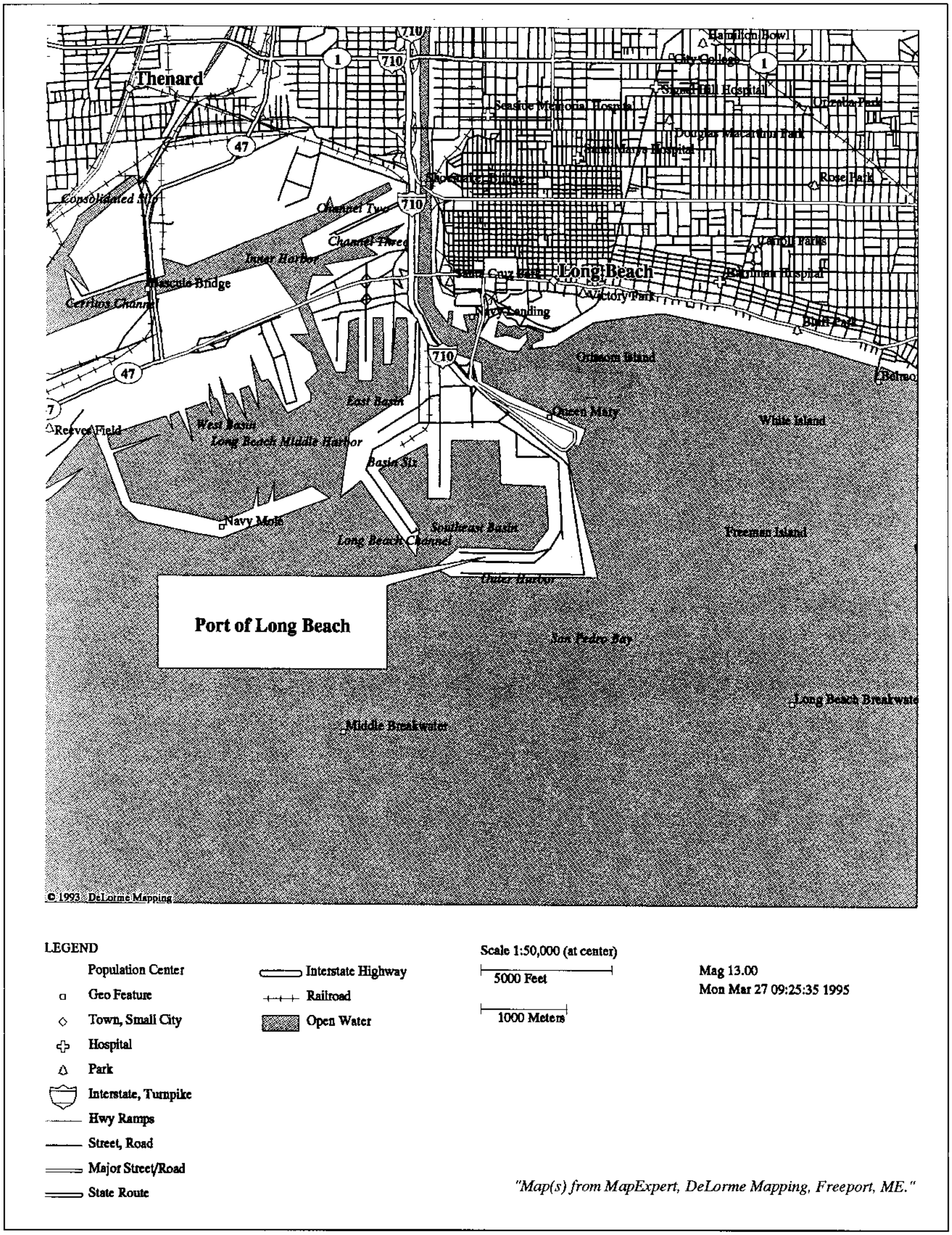


Figure D-41 Map of the Port of Long Beach, CA

Other Pertinent Information: Terminal operators are responsible for the security of their respective facilities. Container terminals are fenced with controlled access and the security forces are port employees (Powell et al., 1994). The port contracts with the City of Long Beach for police and fire protection services. The City of Long Beach stations two fireboats within the port area. There are locations within the terminals for temporary storage of hazardous materials (Hilliard, 1993) but no special areas set aside (Powell et al., 1994).

There are no known environmentally sensitive areas within the harbor area. However, the port claims a long-term interest in maintaining a high quality environment and supports a number of programs to prevent contamination of air and harbor water quality. It was the first recipient of the American Association of Port Authorities Environmental Improvement and Protection Award, and enforces strict safety policies as well. "In the past 50 years, there have been no collisions between commercial vessels resulting in injuries . . . and no significant oil spills from oil transfers." (POLB, 1993b).

The port Marketing Manager did not know of any regulation prohibiting the handling of spent nuclear fuel (Hilliard, 1993). According to available data, the port has not handled spent nuclear fuel since at least 1979 (NRC, 1993; SNL, 1994). The Port of Long Beach does handle other hazardous cargoes and has a number of deep-draft petroleum and petrochemical terminals with attendant tanker traffic, including very large crude carriers. There appears to be good separation between these terminals and the two public container terminals at Pier E and Pier J (POLB, 1993a-d; Jane's, 1992).

Terminal operators contract with private hazardous materials response organizations to contain and control hazardous materials incidents on their premises. The Coast Guard and the Long Beach Fire Department's hazardous materials team are also used for emergency response. Hazardous materials training within the port is the responsibility of the port's Security Division. Port employees receive first responder training for hazardous cargo accidents (no Department of Transportation training), but the Fire Department is the responder for all port accidents (Powell et al., 1994). The Fire Department also calls on the county hazardous materials team as needed (Powell et al., 1994).

The port is physically separated from downtown Long Beach and has excellent highway connections. However, truck and/or rail passage from the terminals must pass through the heart of the adjoining communities that are large and densely populated, which makes the port less than ideal for spent nuclear fuel shipments.

The port is subject to severe earthquakes. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These requirements are shown in the Uniform Building Code (UBC, 1991). For the Port of Long Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.40 g (the highest Uniform Building Code ranking). Nearby San Fernando, CA was the site for one of the worst recorded earthquakes in the contiguous United States with a Modified Mercalli Intensity XI, in February 1971 (Bolt, 1978). Numerous other serious earthquakes with Intensities ranging from IX to X have also occurred in the last century. Long Beach was the site for a Intensity IX earthquake on March 10, 1933, which also resulted in numerous deaths, injuries, and building damage (Bolt, 1978).

The 1990 population within 16 km (10 mi) of the port terminals was 1,014,418. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 914,00; Oak Ridge Reservation, 823,000; Idaho National Engineering Laboratory, 692,000; Hanford Site, 617,000; and Nevada Test Site, 518,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-7 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 3,940 km (2,443 mi); Oak Ridge Reservation, 3,610 km (2,246 mi); Idaho National Engineering Laboratory, 1,580 km (979 mi); Hanford Site, 2,000 km (1,241 mi); and Nevada Test Site, 645 km (401 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The environmental conditions for Long Beach are the same as those for Los Angeles. These are reported in Section D.2.2.11 below.

Climatic Conditions

Similar to the Los Angeles area, the climate of Long Beach, CA, is influenced significantly by the local topography. The Pacific Ocean has a moderating effect on the diurnal temperature range, which is greater than that observed further inland at the Los Angeles International Airport. In general, winter months are cool and wet followed by warm, dry summer months. Early morning clouds and fog, which are quite common during the late evening and early morning hours, generally burn off by late morning, resulting in sunny, pleasant daytime conditions during summer (NOAA, 1993f).

D.2.2.11 Los Angeles, CA

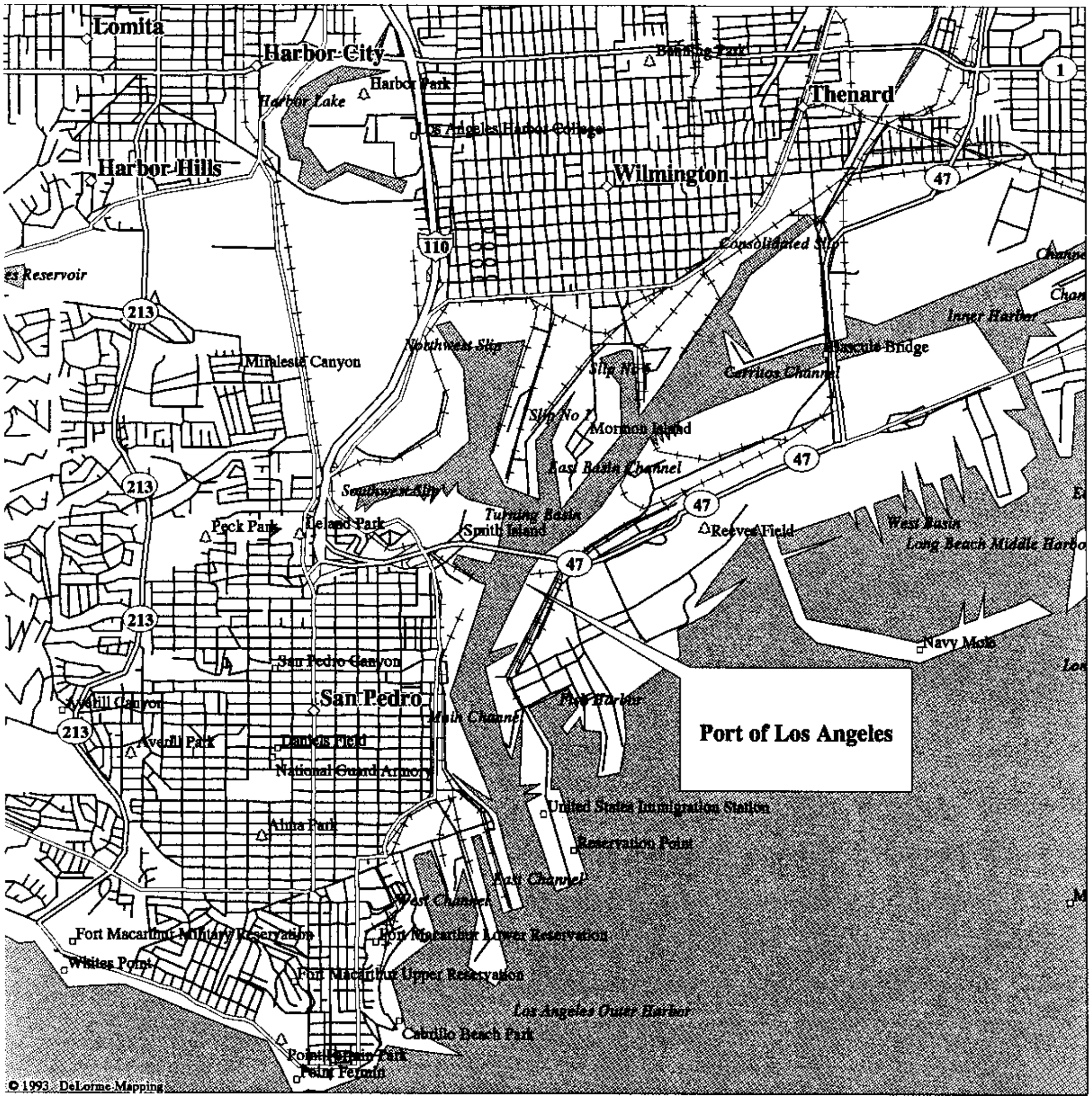
Los Angeles and Long Beach Harbors, although divided by a political boundary, form a single geographic and economic port entity. The harbors occupy a major part of San Pedro Bay. The Port of Los Angeles, one of the largest ports on the Pacific Coast, has a history of leading the Pacific Coast ports in terms of tonnage handled. It has extensive facilities to accommodate all types of traffic, and is the only southern California port at which passenger vessels call regularly (POLA, 1994).

The channel from the Pacific Ocean is straight, short, and direct. The Los Angeles Main Channel is maintained at 13.7 m (45 ft). The Super Tanker Channel to the deep draft facilities is maintained at 12.2 m (40 ft). The majority of the port facilities are located within 4.8 km (3 mi) of the harbor entrance (DOC, 1992b). A map of the port is shown in Figure D-42.

Worldport LA, the name adopted by the Los Angeles Harbor Department for the Port of Los Angeles, is a proprietary and self-supporting department of the City of Los Angeles reporting to a Board of Harbor Commissioners. The Worldport LA functions as a landlord operator administering its own budget, operations, and development programs (POLA, 1994).

Worldport LA is one of the country's largest, multi-terminal ports, and claims the title of the busiest container port in the United States. In fiscal year 1992, Worldport LA handled 2,154,890 20-ft equivalent units — the highest volume in the port's history. The port is also a cruise ship terminus handling over three-quarter million passengers in 1992 (AAPA, 1993; POLA, 1994).

Worldport LA encompasses approximately 1,684 ha (4,160 acres) of land area and 1,425 ha (3,520 acres) of sheltered waters. It has 36 cargo handling terminals, including six dedicated container terminals and three "Omni" terminals (which handle containers and breakbulk) with a total of 34 container cranes, on 45 km (28 mi) of waterfront. Of the three Omni terminals, Berths 142-146 (operated by Worldport LA) is a public facility with no "tenant" ship lines. The remaining two Omni terminals, RDP and Indies Terminals, are managed by private terminal operating companies but are open to public use (Jane's, 1992; AAPA, 1993; D&B, 1993). A multi-billion dollar, outer harbor Pier 300 development is underway with completion scheduled by the year 2010 (some terminals may be open during the period analyzed in this EIS) (POLA, 1994).



LEGEND

- Population Center
- State Route
- Geo Feature
- Town, Small City
- Park
- Interstate, Turnpike
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- Interstate Highway
- Railroad
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:26:31 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-42 Map of the Port of Los Angeles, CA

Worldport LA Berths 142-146: Dock/Quay lengths are 853 m (2,800 ft) and depths alongside at mean low water are 10.67-11.3 m (35-37 ft) at mean low water. Worldport LA berths have crane capacities of three 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

RDP Terminal: Berths 174-181 have lengths of 1,006 m (3,300 ft) and depths of 10.67 m (35 ft) at mean low water. Capacity of RDP cranes is two 40.6 metric ton container cranes (Jane's, 1992; AAPA, 1993).

Indies Terminal: Berths 216-227 have total lengths of 1,128 m (3,700 ft) and depths of 13.72 m (45 ft) at mean low water. The Indies berths have three 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

Los Angeles terminals are served by the Harbor Freeway (I-110) and Terminal Island Freeway (Route 47) which connect with Interstate Highways 5, 10, 15, and 40. The Harbor Freeway begins within the Worldport LA port complex. Worldport LA is connected to the Southern Pacific Transportation Co., Union Pacific, and Santa Fe railroads by the Harbor Belt Line Railroad, jointly owned by the Los Angeles Harbor Department and the three railroads. Belt Line tracks extend to cargo ship berths at each of the Omni Terminals. Intermodal connections are presently made at the intermodal container transfer facility described for the Port of Long Beach, which is approximately 8 km (5 mi) away. A new intermodal container transfer facility is under construction on Terminal Island and there are major infrastructure improvement projects underway to facilitate and expedite rail and truck traffic to the port through the Greater Los Angeles Metropolitan area (POLA,1994).

Worldport LA is host to more than a dozen cruise ship lines and about 40 cargo ship lines. A partial list of container lines calling at the port include: American President Lines, Australia-New Zealand Direct Line, Orient Overseas Container Line, Philippines, Micronesia & Orient Lines, Yang Ming Line, Mitsui O.S.K. Lines, Kawasaki Kisen Kaisha ("K" Line), Dole Fresh Fruit, Columbus Line, Blue Star PACE Ltd., Matson Navigation Co, NYK Line, Neptune Orient Lines, Evergreen Line, Barber Wilhelmsen, Blue Star Line, d'Amico Line, Italia Line, Nedlloyd Lines, and Slosna Plovba (Jane's, 1992; D&B, 1993).

Other Pertinent Information: The Port of Los Angeles has its own police force that patrols the waterfront around the clock by boat, helicopter, automobile, and bicycle (Leong, 1993). Port security is extensive and extremely well-organized. The port police are responsible for the safety and security of all passenger, cargo, and vessel operations at Worldport LA. They also monitor vessel berthings for possible wharf damage and issue hazardous cargo and dangerous goods permits. In addition, terminal operators have their own unarmed security personnel. All terminals also have areas for segregation and temporary storage of dangerous cargoes (Verhoef et al., 1994).

Fire protection is provided by the Los Angeles Fire Department which maintains five fire stations within the port and operates five fireboats. Two additional fireboat berths and stations are under construction. Response time is within five minutes. First responders for accidents receive Occupational Health and Safety Administration training but do not yet receive Department of Transportation training (Leong, 1993; Verhoef et al., 1994).

A port spokesperson did not know of any ordinances prohibiting the importation of spent nuclear fuel (Leong, 1993). A port spokesperson thought the port had handled spent nuclear fuel shipments in the past (Note: Database searches of shipments over the last decade do not show Los Angeles as a port for receipt of spent nuclear fuel; presumably these past shipments were other types of radioactive materials) (Leong, 1993). Item 1715 of Los Angeles Port Tariff No. 4, effective July 1, 1990, provides for the handling of radioactive and/or fissile materials, provided special written permission is received from the Executive

Director and U.S. Department of Transportation/Coast Guard Regulations are fully complied with (POLA, 1994). However, a spokesperson indicated that it was unlikely the port would accept spent nuclear fuel shipments (Verhoef et al., 1994).

The port police are the primary responders to hazardous materials incidents, backed up by the Los Angeles Fire Department and the United States Coast Guard. Based on Tariff Item 1715 referred to above, and the fact that radioactive shipments have occurred in the past, it is assumed port police have an adequate handling plan in place for radioactive materials. Worldport LA is an active participant in the Shoreline Emergency Network, a regional oil spill network organized to respond to coastal oil spill emergencies. The port police are trained in hazardous materials handling and are in charge of such operations. It is not known to what extent individual terminal operators are trained in hazardous materials response, but given the size and complexity of the port activities, it is assumed adequate hazardous materials training is provided. The combined ship accident history for the Ports of Long Beach and Los Angeles for the period of 1991-1993 is the lowest of all the major west coast ports (USCG, 1994b).

Worldport LA has a number of environmental programs underway that are designed to mitigate damage done to the marine environment in the past, and to prevent or lessen additional negative environmental impacts in the future. The port has a very active recreational/tourist component and, due in part to the presence of oil production facilities within the port, there is heightened environmental sensitivity on the part of the port community. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Los Angeles, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.45 g. Like most Southern California cities, the port is subject to severe earthquakes. Two relatively recent severe earthquakes in Southern California (along the San Andreas fault system along the Pacific and North American tectonic plates) occurred March 10, 1993, in Long Beach (Modified Mercalli Intensity IX) and February 9, 1971, in nearby San Fernando (Modified Mercalli Intensity VIII-XI). Both resulted in numerous deaths and injuries and caused massive structural damage to buildings.

The 1990 population within 16 km (10 mi) of the port terminals was 1,124,493. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 936,000; Oak Ridge Reservation, 639,000; Idaho National Engineering Laboratory, 519,000; Hanford Site, 725,000; and Nevada Test Site, 334,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 3,930 km (2,440 mi); Oak Ridge Reservation, 3,550 km (2,204 mi); Idaho National Engineering Laboratory, 1,510 km (940 mi); Hanford Site, 2,070 km (1,286 mi); and Nevada Test Site, 580 km (359 mi). Distances along rail routes are slightly longer.

Environmental Conditions

Several wildlife refuges are located around the San Pedro Bay area. The Seal Beach National Wildlife Refuge is approximately 16 km (10 mi) east from the port area. The Bolsa Chica Ecological Reserve is located about 20 km (11 mi) east from the port area. The Abalone Cove Ecological Reserve is about 16 km (10 mi) around Point Fermin to the west of the port area. Several areas of biological significance lie along the passageway to the ports. These include the Channel Islands National Marine Sanctuary, about 60 km (43 mi) to the southwest of the port entrance, and the Santa Catalina Island Area of Special Biological Significance, about 28 km (17 mi) southwest of the port entrance (FWS, 1981c).

The water quality of the harbor waters at the ports is generally considered good. Areas in the inner harbor with limited waterflow experience infrequent periods of poor water quality. There is a wider range of salinity in the inner harbor than in the outer harbor, with higher values at the bottom than at the surface (U.S. Army, 1990).

The waters of the Los Angeles-Long Beach Harbor contain a variety of marine habitats, some natural and some manmade. Numerous fish species use the habitats in the harbor, including several recreational (e.g., barred sand bass and white croaker) and commercial (e.g., anchovy and halibut) value, for all or part of their life cycle (U.S. Army, 1990). Commercial fishing operations for crabs and spiny lobsters also are in San Pedro Bay. Other sport-fishing in the bay includes flatfish, grunions, California halibut, white seabass, kelp bass, Pacific bonito, and Pacific barracuda (FWS, 1981c). Shallow waters are important nursery areas for several fish species. At least 60 species of water-associated birds use the harbor, primarily for resting and foraging (U.S. Army, 1990).

Several threatened or endangered species are present at least seasonally in San Pedro Bay (Kobetich, 1994; U.S. Army, 1990). The endangered California least tern breeds in the area from April through August. California brown pelicans are present all year feeding on the fish in the harbor and resting on the breakwaters and other structures. Peregrine falcons are present in the region but are seldom sighted in the harbor. Other endangered birds around the ports include the light-footed clapper rail and the marbled morrelat. The western snowy plover (threatened) and the long-billed curlew, which is a candidate species, have been spotted infrequently in the harbor. Other candidate species, including the elegant tern, harlequin duck, loggerhead shrike, reddish egret, and white-faced ibis, can be found in the harbor area.

Within the Seal Beach National Wildlife Refuge, the wetlands portion supports a wide variety of fish and invertebrate as well as residential and migratory bird populations. The bay provides habitat for the light-footed clapper rail, Belding's savannah sparrow, the California least tern and the California brown pelican, all of which are endangered (U.S. Army, 1990). The Bolsa Chica Ecological Reserve provides habitat for the California least tern, the light-footed clapper rail, the California brown pelican, Belding's savannah sparrow (State protected), and the salt marsh bird's-beak, a member of the figwort family. The Belding's savannah sparrow is strictly associated with pickleweed, which is not found within the ports; therefore this species is not expected to be found directly in the ports. The reserve is also used by the coast horned lizard, monarch butterfly, snowy plover, and numerous bird species such as gulls, terns, sandpipers, herons, and egrets.

With regard to marine mammals, no species of cetaceans (whales, dolphins) actually inhabit regions in-shore of the breakwater, and their occurrence within the harbor is sporadic and infrequent. Visitors include the common dolphin, the Pacific white-sided dolphin, and gray whale (endangered). Groups of bottlenose dolphins have been observed swimming just outside the breakwater. The eastern Pacific gray whale migrates through California waters twice yearly in a route between the Bering Sea and Baja California. The southward migration occurs between November and February, while the northward return generally takes place off of California between March and May. While the gray whales usually stay outside the harbor mouth, approximately three to four accidently enter the harbor every year. The California sea lion and the harbor seal, both nonendangered, have been sighted in the area of the harbor. The California sea lion is known to occasionally haul-out on the harbor breakwater and sometimes can be seen swimming in the harbor. The harbor is not considered a birthing or important feeding habitat for the California sea lion, although sea lions could presumably forage within the harbor (U.S. Army, 1990).

Climatic Conditions

The dominant geographic influences on the climate of the Los Angeles basin are the Pacific Ocean and the southern California coastal mountain ranges. Marine air covers the coastal plain for the majority of the year, but inland air does occasionally migrate into the region. Pronounced differences in temperature, humidity, cloudiness, fog, sunshine, and rain occur over fairly short distances along the coastal plain due to the local topography and the decreased effect of the marine environment further inland. However, in general, temperature ranges are least and humidity highest close to the coast, while precipitation increases with elevation in the foothills. Prevailing daytime winds are from the west, with nighttime and early morning winds generally light and from the east and northeast. During the fall, winter and spring months, dry, gusty northeasterly winds (e.g., Santa Ana winds) blow over the southern California mountains. Precipitation occurs mainly during the winter months. Thunderstorms are rare along the coast, but increase in frequency as one approaches the coastal ranges. Fog and low visibility are frequent problems for aircraft navigation at the Los Angeles International Airport (NOAA, 1993e).

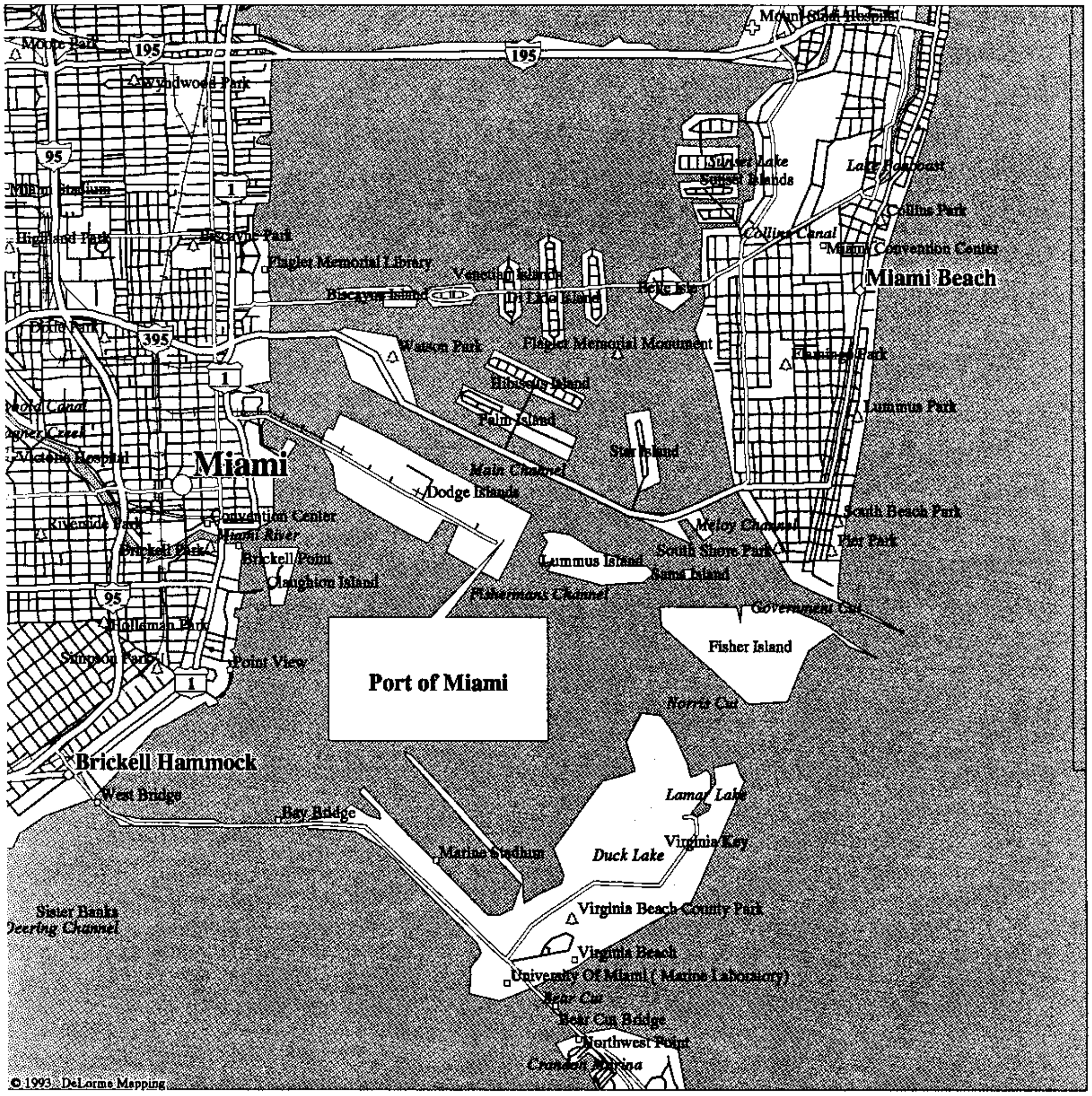
D.2.2.12 Miami, FL

Miami is Florida's most populous city and is located 8 km (5 mi) from the Gulf Stream on the east coast of Florida. It is an internationally famous winter resort and a popular yachting center. Miami is also a deepwater port; considerable foreign commerce passes through Miami and it is a major cruise port. Miami's cruise ship traffic has earned it the title of "Cruise Ship Capital of the World" (Southern Shipper, 1993). In addition to being a major shipping and cruise ship center, the Port of Miami is located in a popular resort area known for its beaches, fishing, recreational boating, and tropical landscape. The approach to Miami is open, but with strong tidal currents of 1.0 to 2.1 meters-per-sec (2 to 4 knots) in the entrance between the jetties. A Federal project provides for depths of 11 m (36 ft) to the main port facilities (DOC, 1993d; Southern Shipper, 1993; AAPA, 1993; Jane's, 1992). The port occupies 273 ha (675 acres) of land. It is situated on two interconnected islands, Dodge and Lummus, which lie in an east-west orientation due east of the City of Miami and west of the barrier island resort area of Miami Beach. The Miami Beach resort area forms the northern boundary of the harbor entrance. The major port facilities are within 5 km (3 mi) of the entrance from the Atlantic Ocean. A map of the port is shown in Figure D-43.

Miami's freight terminals serve as a hub for distribution and transshipment of cargo (largely tropical fruits and vegetables) to and from Latin America. The Port of Miami is an arm of the Dade County Seaport Department which functions as a "landlord" port. Almost 60 shiplines connect the port to most major countries in the world (Jane's, 1992; D&B, 1993; Southern Shipper, 1993). In 1994, nearly 520,000 20-ft equivalent-units were handled in the port (AAPA, 1994).

Lummus Island Terminal: The 91 ha (225 acre) terminal on the south side of the island is seaward of Dodge Island and just inside the entrance to the port. It is Miami's principal container handling facility with six container gantry cranes, including three new post-Panamax cranes and a new roll-on/roll-off berth. Activities at Dodge Island are primarily cruise ship, roll-on/roll-off, and breakbulk cargo oriented. Combined facilities consist of four container berths, 14 roll-on/roll-off berths, and 12 cruise ship berths. A private container terminal for shallow draft vessels is located on Causeway Island at the eastern end of the MacArthur Causeway, which parallels the ship channel north of Lummus and Dodge Islands (Southern Shipper, 1993).

This terminal has marginal wharf area of 1,067 m (3,500 ft). The roll-on/roll-off berths have 413 m (1,356 ft) of marginal wharf while the Dodge Island breakbulk has 853 m (2,800 ft) of marginal wharf. The passenger terminals have 2,373 m (7,785 ft) of marginal wharf. Depths alongside Dock/Quay are



LEGEND

- Population Center
- State Route
- Geo Feature
- Park
- Hwy Ramps
- Street, Road
- Major Street/Road
- State Route
- US Highway
- Railroad
- River
- Land Mass
- Open Water

Scale 1:50,000 (at center)

5000 Feet

1000 Meters

Mag 13.00
Mon Mar 27 09:28:12 1995

"Map(s) from MapExpert, DeLorme Mapping, Freeport, ME."

Figure D-43 Map of the Port of Miami, FL

noted as: Lummus Container Berths 1 and 2 with 12.8 m (42 ft) at mean low water. Berths 3 and 4 have 11.73 m (38.5 ft) at mean low water. On the north side of Dodge Island, the depth is 7.62-10.97 m (25-36 ft) at mean low water.

Crane capacities at Lummus Container Berths consist of three 50.8 metric ton (56 ton) container gantry cranes and three 40.6 metric ton (45 ton) container gantry cranes.

The Port of Miami is accessible via a five-lane, fixed bridge spanning the Intracoastal Waterway. It is approximately 1.2 km (0.75 mi) from the Biscayne Boulevard exit of I-395 to Dodge Island via NE 2nd Avenue in downtown Miami. I-395 is a connector to I-95 as well as all other south Florida highways. There are 5.2 km (3.2 mi) of trackage within the Port of Miami including a four-track marshalling yard. Rail connections are with the Florida East Coast and CSX Railroads.

Port users include Agromar, Argentine Line, Barber Blue Sea, Bottachi Line, Central American Shippers, CCNI, CGM, Chilean Line, Ecuadorian Line, Empremar, Flota Mercante Grancolombiana, Hapag Lloyd, Hoegh Line, Italian Lines, Ivaran Lines, Kirk Line, Lykes Line, Maersk Line, Navieras De Puerto Rico, Shipping Corp of India Transnave, Mexican Line, Spanish Line, Wallenius Transroll, and Zim Container Service (Jane's, 1992; Southern Shipper, 1993; D&B, 1993).

Other Pertinent Information: Containers discharged at Lummus Terminal must travel down the center of both islands and past the extensive cruise line terminals located on the north side of Dodge Island. Although travel on city streets on the mainland is for a very short distance, it is through an urban area which is believed to be heavily developed. The port recently completed a new $1.8 million, eight-lane security gate and cargo control facility on Dodge Island to the east of the bridge. Each lane is equipped with Regiscope photographic clearance systems (Southern Shipper, 1993).

Port officials did not respond to a faxed questionnaire or telephone calls for information, and it is not known if there is a designated area for temporary storage of hazardous cargoes. The port has no prior experience handling spent nuclear fuel (NRC, 1993; SNL, 1994). However, the Port of Miami is primarily a general cargo, container, and cruise ship port with no petroleum berths or other terminals for handling hazardous or dangerous goods. Passenger operations are considered a conflicting use. Since port officials did not respond to requests for information regarding emergency response capabilities, it is not known whether hazardous materials or spent nuclear fuel training exists for port workers.

There are no known wildlife habitats or sanctuaries in the immediate area; however, there is a high-level of environmental sensitivity in this area. The port's physically separate island locations, strictly controlled access, and limited use of city thoroughfares are very desirable features. However, the port is in relatively close proximity to the heavily populated Miami Beach area adjoining the harbor entrance (Government Cut) and roughly 0.8 km (0.5 mi) from downtown Miami.

The likelihood of severe natural phenomena, such as high winds and earthquakes, are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Miami, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 833,057. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 336,000; Oak Ridge Reservation, 443,000; Idaho National Engineering Laboratory, 845,000; Hanford Site, 894,000; and Nevada Test Site, 908,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,200 km (748 mi); Oak Ridge Reservation, 1,460 km (906 mi); Idaho National Engineering Laboratory, 4,570 km (2,842 mi); Hanford Site, 5,240 km (3,258 mi); and Nevada Test Site, 4,740 km (2,945 mi). Distances along rail routes are slightly longer, except for Savannah River Site, which is slightly less.

Environmental Conditions

The State of Florida has classified Biscayne Bay near Port of Miami as a Class III water body. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). In addition, the State of Florida has classified the Biscayne Bay, where the Port of Miami is located, as an "Outstanding Florida Waterway." As previously noted, Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Other waters located near Port Everglades that are designated as Outstanding Florida Waters include Biscayne Bay National Park and the Bill Bags State Recreation Area. These waterways are afforded special protection by State environmental regulations (FL DEP, 1994).

The Biscayne Bay, in the vicinity of the Port of Miami, is characterized as a high salinity estuarine habitat (generally greater than 20 parts per thousand). There are both commercial and recreational fish and invertebrates found near the port. These aquatic species include: stone crabs, shrimp, spiny lobster, sharks, sand seatrout, drum, kingfish, mullet, Florida pompano, bluefish, mackerel, tarpon, ladyfish, snapper, grouper, grunts, jewfish, snook, greater amberjack, crevalle jack, silver perch, blue runner, Atlantic dolphin, short-finned pilot whale, false killer whale, and pygmy sperm whale (FWS, 1982c).

Protected species found near the Port of Miami include the loggerhead sea turtle and the West Indian manatee (Richards, 1994). The U.S. Fish and Wildlife Service reported that the Port of Miami is located in designated critical habitat for the endangered west indian manatee (Johnson, 1995). In addition, the U.S. Fish and Wildlife Service reports that the following protected, listed marine species are known to occur in Dade County: atlantic hawksbill turtle (endangered), atlantic ridley turtle (endangered), atlantic loggerhead turtle (threatened), atlantic green turtle (endangered), leatherback turtle (endangered), american crocodile (endangered), and the american alligator (threatened/similar appearance). Protected bird species in Dade County include the bald eagle (endangered), cape sable seaside sparrow (endangered), ivory-billed woodpecker (endangered), kirtland's warbler (endangered), arctic peregrine falcon (threatened), wood stork (endangered), everglades snail kite (endangered), bachman's warbler (endangered), roseate tern (threatened), and the piping plover (threatened) (Johnson, 1995).

Wildlife refuges located near the port area are the Bill Baggs Cape Florida State Recreation Area and the Biscayne Bay Aquatic Preserve. They are both located within 20 km (12 mi) of the Port of Miami. Protected species found in these areas include the loggerhead sea turtle, the West Indian manatee, and the peregrine falcon. Birds of interest found in these areas are: the spotted breasted oriole, songbirds, fulvuous whistling duck, and various shorebirds (FWS, 1982c).

Climatic Conditions

The climate of the southeast Florida region is essentially subtropical marine, which features long, warm summers with abundant rainfall, generally followed by a mild, dry winter. The influence of the ocean and numerous bays is seen in the small diurnal temperature range (generally <10°) and the rapid warming of any cold air masses that invade this portion of the State. The predominant windflow is from the east-southeast, which generates conditions right at the coast that are often different than those encountered further inland, due to land-induced frictional effects. Hurricanes occasionally effect the area, with the months of September and October exhibiting the highest frequencies. However, destructive tornadoes (not associated with tropical systems) are rare. Waterspouts are frequently spotted offshore during the summer months, but rarely cause any loss of life or property damage (NOAA, 1993b).

D.2.2.13 Military Ocean Terminal, Oakland, CA

The Military Ocean Terminal, Bay Area, is located in the Outer Harbor of the Port of Oakland, adjacent to the east entrance to the Oakland Bay Bridge (descriptions of Oakland ship channels also apply to Military Ocean Terminal, Bay Area and are not repeated here). The facility is located approximately 16 km (10 mi) east of the Golden Gate Bridge, which spans the Pacific Ocean entrance to San Francisco Bay to the south and San Pablo Bay to the north. The single pier (Wharf 7) currently available for military cargo is directly opposite the commercial Sea-Land and Public Container Terminals, and located within the Oakland Army Base (MTMCTEA, 1990). The facility has the largest sealift workload of any military traffic ports on the West Coast, averaging on the order of 3,000 metric tons (3,300 tons) of cargo per year (the 1994 shipments of Patriot missiles to South Korea were shipped from Wharf 7). See the descriptions of the Port of Oakland for more information regarding truck and rail access, maps, populations, etc. A map of the terminal is shown in Figure D-44.

The Bay Bridge Terminal, adjacent to Military Ocean Terminal, Bay Area, operates Military Ocean Terminal, Bay Area wharves 6 and 6.5 as Berths 8 and 9 for its commercial operation (Jane's, 1992; MTMCTEA, 1990). Wharf 7 is 445 m (1,459 ft) long, with 10.6 m (35 ft) depth alongside. Wharf 7 has a single 91 metric ton gantry crane for all breakbulk operations and a container spreader that can be attached for limited container handling (MTMCTEA, 1990). A floating crane of comparable capacity is also available. Stern loading roll-on/roll-off operations are not feasible at the wharf.

There are more than 8.1 ha (20 acres) of open storage space near the wharf, and a transit shed at the wharf provides more than 13,000 m^2 (141,000 ft^2) of covered storage. More than 65,000 m^2 (700,000 ft^2) of additional covered space is available on the Army Base (MTMCTEA, 1990).

Trucks can access the wharf for direct loading from ships at the facility. Access to Interstates 580, 680, or 880 is directly adjacent the Army Base through a largely industrial area at the Port of Oakland. Residential areas are within a few kilometers of the Base and the Port of Oakland.

The entire length of Wharf 7 is served by rail, making direct ship-to-rail loading possible for receipt of incoming cargo. Rail movements are carried out by two Base locomotives, which can move rail shipment to the adjacent and expanding Oakland Intermodal Terminal. The Intermodal Terminal is serviced by the Southern Pacific and Union Pacific rail systems and connections with the Atchison, Topeka, and Santa Fe Railroad intermodal yard about 19 km (12 mi) north of the port (MTMCTEA, 1990).

Other Pertinent Information: Since the facility is part of the Oakland Army Base, it is well lighted, fenced, and patrolled by gate guards and roving patrols. There are no full time longshoremen at the facility, and trained, experienced longshoremen are hired from the large pool of stevedores (1,000) normally working at the port.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Military Ocean Terminal, Bay Area, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). The port is located in a very high seismic zone with an acceleration of 0.40 g (see seismic information for the Port of Oakland for more details).

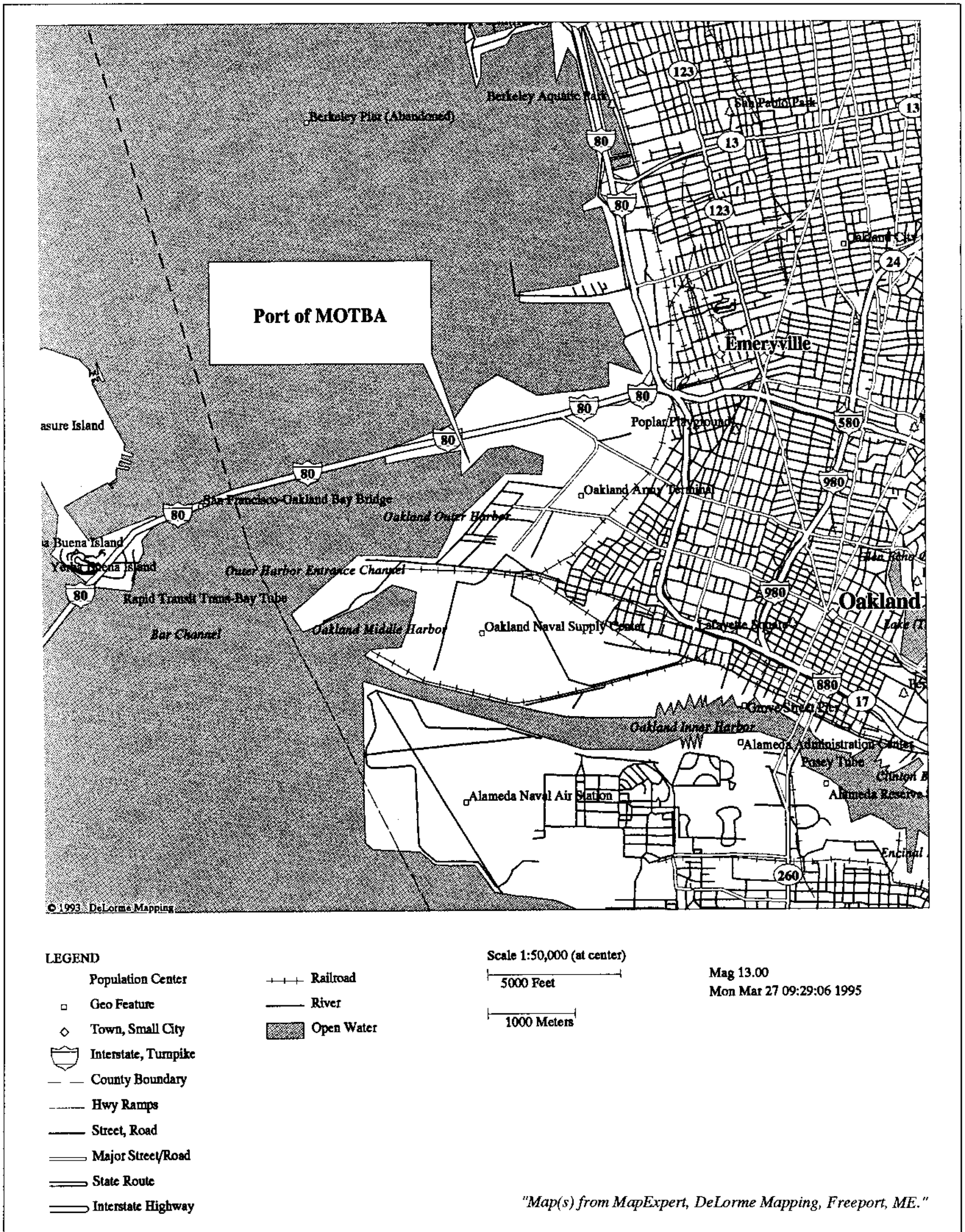


Figure D-44 Map of the Military Ocean Terminal, Oakland, CA

Area 1990 census population and density figures are 1,110,549 and 1,323 persons/km^2 (511 persons/mi^2), respectively. The 1990 population within 16 km (10 mi) of the port terminals was 1,288,899. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 786,000; Idaho National Engineering Laboratory, 367,000; Hanford Site, 359,000; and Nevada Test Site, 482,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,216 km (2,791 mi); Oak Ridge Reservation, 4,121 km (2,563 mi); Idaho National Engineering Laboratory, 1,548 km (963 mi); Hanford Site, 1,407 km (875 mi); and Nevada Test Site, 1,156 km (719 mi). Distances along rail routes are slightly longer.

The Military Ocean Terminal, Bay Area, is located in the Outer Harbor of the Port of Oakland. Climatic and environmental conditions for Military Ocean Terminal, Bay Area are the same as those for the Port of Oakland. These are presented in Section D.2.2.15.

D.2.2.14 New Orleans, LA

The Port of New Orleans is one of the largest ports in the United States. It is located on both sides of the Mississippi River with its lower limit about 129 km (80 mi) above the Head of the Passes from the Gulf of Mexico, and its upper limit about 185 km (115 mi) above Head of the Passes. A Federal project provides for a channel 13.7 m (45 ft) deep over the bar through Southwest Pass to Head of the Passes, and on to New Orleans. The Port of New Orleans' lower limit is about 160 km (98 mi) from the Gulf of Mexico via Southwest Pass. Southwest Pass is straight and well-marked. From the Head of the Passes to New Orleans, the river has a least width of 550 m (1,800 ft) and a clear, unobstructed channel (DOC, 1992a). A map of the port is shown in Figure D-45.

The seven-person Board of Commissioners of the Port of New Orleans, is appointed by the Governor from a list of nominees drawn from industry, civic, and educational groups from the three parishes (counties) in which the Port of New Orleans' terminals are located. The Board, a state agency, sets policies and regulations for port operations. It also appoints the president and chief executive officer of the Port of New Orleans who, together with a staff of professional managers, are responsible for day-to-day operation of the port.

New Orleans is a multi-terminal port with predominantly publicly owned terminals and a few private terminals. The port is strictly a "landlord" operator, leasing all of its terminals to private operators and/or shipping companies. Most of the large publicly owned terminals are located along the banks of the Mississippi (on the New Orleans side of the River), which generally runs in an east-west direction in the vicinity of the City. In 1994, the port handled over 250,00 20-ft equivalent units of containerized cargo (AAPA, 1994).

France Road Container Terminal is the Port of New Orleans' principal container handling facility. It occupies 71.55 ha (177 acres) of land and is situated on the west bank of the industrial canal in the southwestern section of New Orleans at the intersection of the industrial canal with the Mississippi River/Gulf Outlet. Berths 1 and 4 are leased to Sea-Land and Navieras De Puerto Rico respectively, and Berths 5 and 6 are public terminals. Berth 1 has two 30.5 metric ton (34 ton) container cranes. Berths 5 and 6 are supported by three container cranes [one 30.5 metric ton (34 ton) container crane and two 40.64 metric ton (45 ton) container cranes], a marshalling yard of 195,077 m^2 (2.1 million ft^2), two container freight stations with 12,193 m^2 (131,120 ft^2) of consolidating space, and a roll-on/roll-off ramp

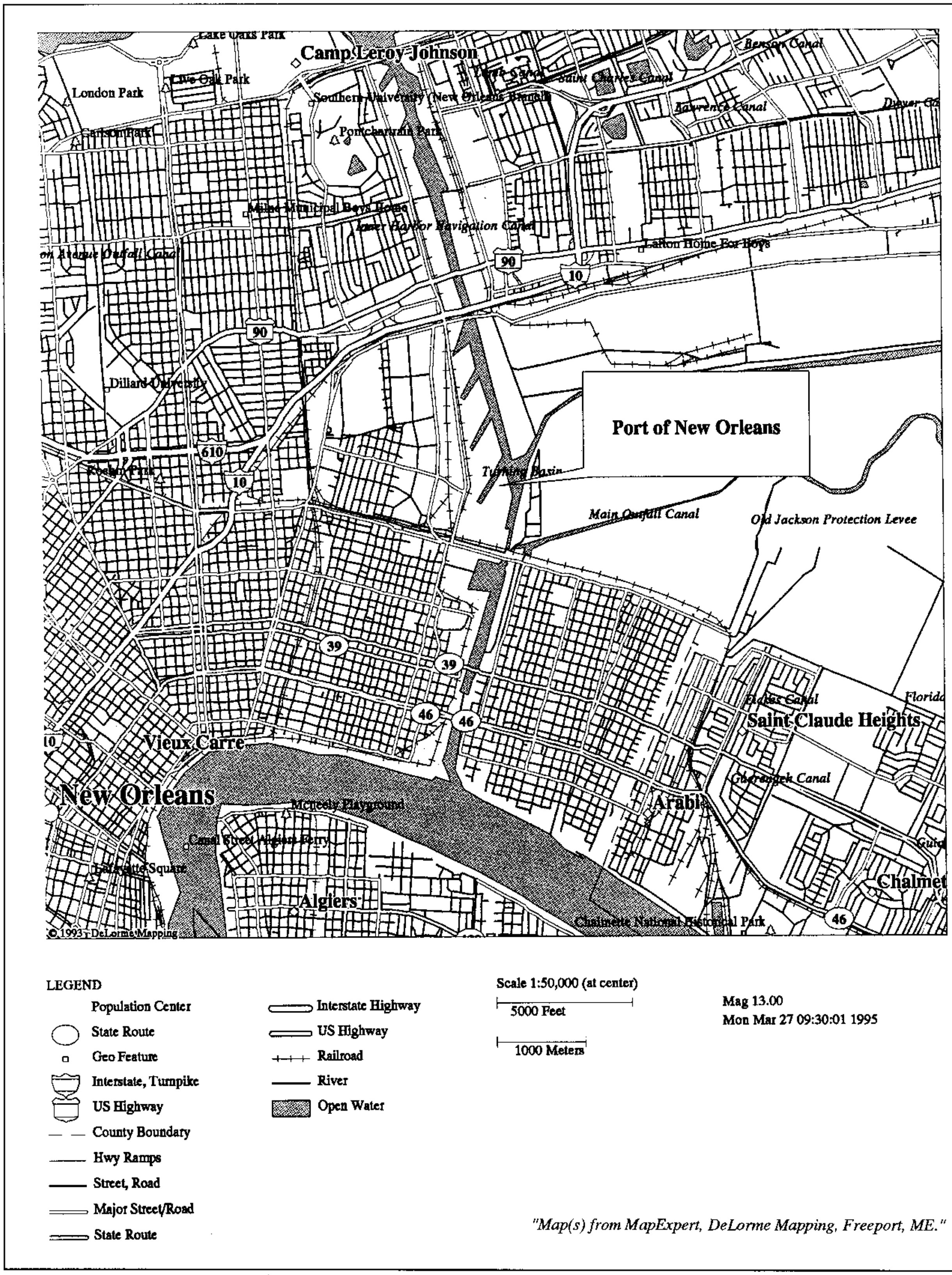


Figure D-45 Map of the Port of New Orleans, LA

at Berth 6. Berths 5 and 6 have a combined length of 518 m (1,700 ft) of marginal wharf with water depth alongside ranging from 9.75 - 10.97 m (32 - 36 ft) at mean low water (AAPA, 1993; Janes's, 1992; D&B, 1993; Southern Shipper, 1993; PON, 1994).

The France Road Public Container Terminal is located in a port industrial district that appears to be separate from residential areas and a considerable distance from the downtown New Orleans business district. This terminal is about 2.4 km (1.5 mi) from Interstate Highway 10 and U.S. Route 90 — major east-west highways — via Alvar Street or Florida Avenue, which are heavy truck routes. I-10 connects with I-49 to Shreveport, where it meets I-20. The Terminal has good truck and rail access, but waterway access is via the relatively narrow industrial canal with a lock near the entrance and several bridges en route. The city-owned Public Belt Railroad connects the France Road and other terminals on the Mississippi, Industrial Canal, and Mississippi/Gulf Outlet with the CSX, Illinois Central, Kansas City Southern, Norfolk Southern, Southern and Union Pacific Railroads. In the case of France Road Terminal, the Belt Railroad tracks serve the site, but not the pier apron.

The Port of New Orleans is port-of-call for over 50 steamship lines providing breakbulk and container freight service to virtually all of the world's major port cities. A partial list of these lines includes ABC Container Line, Argentine Line, ART Ocean Line, Atlantic Container Line, Baltic Shipping, Bank Line East Africa, Boss Line, China Ocean Shipping Co., Contship Container Line, Chilean Line, Daiichi-Chuo Shipping Line, Delmas/AAEL, Egyptian National, Forest Lines Inc., Hapag-Lloyd, Hoegh Line, Hyundai Merchant Marine, Industrial Maritime Carriers, Italia Line, Lykes Brothers Steamship Co., Maersk Inc., NCSCA, Pakistan National Shipping, Pan Ocean, Safbank, Sea-Land, Tecomar, Toko, Torm West Africa, Turkish Cargo Lines, United Arab Shipping, Waterman/LASH, Wilhelmsen Line, and Zim Container Line (Jane's, 1992; D&B, 1993).

Other Pertinent Information: The port has its own security force with police powers. The France Road Terminal is secured by fencing and controlled access. There are locations within the terminal for isolation of hazardous materials. The Port Harbor Police are the first line of defense with respect to hazardous materials accidents, followed by the Coast Guard and Louisiana State Police, who have primary responsibility for enforcing Department of Transportation Regulations. The port has an "elaborate" notification system in case of accidents on Port Authority terminals, beginning with the shipper or consignee of the goods. The New Orleans Fire Department also has a hazardous materials team. Hazardous materials training is the responsibility of the individual terminal operating companies. While the level of training at each terminal is uncertain, given the large quantities of hazardous materials passing through the port, some training is certain (Parker, Spalluto, and Cefalu, 1993).

Port officials know of no ordinances or regulations prohibiting the importation of spent nuclear fuel through the Port of New Orleans, and thought the port may have handled spent nuclear fuel in the past. However, other data indicate the port has not handled spent nuclear fuel since at least 1979 (NRC, 1993; SNL, 1994). The port spokesperson indicated that shipments of radioactive nuclear fuel (not spent nuclear fuel) have been shipped through the port and may still be coming in. Apparently these shipments were from South Africa (Parker, Spalluto, and Cefalu, 1993).

The Port of New Orleans is primarily a breakbulk and general cargo/container port. It is also a major river barge terminus for barge lines on the Mississippi River system. Although there is considerable tank ship and barge traffic on the River, tank terminals tend to be located on the opposite side of the river and/or outside the City limits. Conflicting use of the waterway is not considered a major factor with regard to handling spent nuclear fuel. However, a U.S. Coast Guard accident database established in 1990 shows that an extremely high number of accidents occur on the transit from the Gulf to the port (USCG, 1994a). During the period 1991 through the third quarter of 1993, there were 790 collisions, 825 allisions, and

1,065 hard groundings reported (see 46 CFR 4.05-1 for reporting requirements and definitions). The 2,680 accidents involved one of the following: vessel damage in excess of $25,000 and/or left the vessel unseaworthy, or without power or steering, or severe injury or death. The port 1993-1994 Annual Directory indicates that during this period, there were about 7,100 vessel transits (PON, 1994). Since the accident statistics include barge accidents (and New Orleans has large barge traffic), this number is rather high for oceangoing vessels, but no data are yet available yet to refine the information.

Other than flooding from severe hurricanes and tropical storms, and general environmental concerns, there are no known special environmental or wildlife issues in or near the port area. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of New Orleans, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 782,868. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 266,000; Oak Ridge Reservation, 256,000; Idaho National Engineering Laboratory, 455,000; Hanford Site, 504,000; and Nevada Test Site, 687,000. Populations along rail routes to these sites are slightly smaller for Idaho National Engineering Laboratory, Hanford Site, and Nevada Test Site and much larger for Savannah River Site and Oak Ridge Reservation. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,020 km (634 mi); Oak Ridge Reservation, 960 km (594 mi); Idaho National Engineering Laboratory, 3,510 km (2,184 mi); Hanford Site, 4,180 km (2,600 mi); and Nevada Test Site, 3,450 km (2,145 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The State of Louisiana has classified the waters of the Industrial Canal and the Mississippi River Gulf Outlet as suitable for primary and secondary water recreational activities and the propagation of fish and wildlife. The Mississippi River supports all of these uses in addition to being used as a drinking water supply source (Fabens, 1994).

The Mississippi River Gulf Outlet, in the vicinity of the France Road Terminal, is characterized as a high salinity estuarine habitat (generally greater than 20 parts per thousand). The entire canal travels through marshlands. Aquatic species found in these types of marshlands and surrounding areas in Louisiana include: shrimp, blue crab, eastern oyster seatrout, Atlantic croaker, drum, spot, kingfish, sheepshead, flounder, mullet, sea catfish, gulf menhaden, bay anchovy, crevalle jack, and Atlantic bottlenose dolphin (FWS, 1982d).

As ships approach the Mississippi River Gulf Outlet from the north they must travel past the Breton National Wildlife Refuge and Breton Wilderness. Birds of interest in these areas include: peregrine falcon, brown pelican, shorebirds, wading birds, herons, egrets, white ibis, least bittern, gallinules, waterfowl, bird hawks, osprey, magnificent frigate-bird, white pelican, songbirds, warblers and diving ducks. The peregrine falcon and brown pelican are protected species. Aquatic species found in these areas include: loggerhead sea turtle, spotted sea trout, drum, bluefish, cobia, and mackerel. The loggerhead sea turtle is a Federally protected species (FWS, 1982d). Travelling north into the Mississippi Gulf River Outlet towards the France Road Terminal ships must pass near the Biloxi Wildlife Management area.

Climatic Conditions

The city of New Orleans is essentially surrounded by water. Thus, the influence of the Gulf of Mexico and the surrounding bayous, lakes, and marshes are significant. The climate can best be described as humid, with the surrounding water significantly reducing the diurnal temperature range. Between mid-June and mid-September, almost daily, sporadic thunderstorms occur and prevent the temperature from rising much above 90°F. From mid-November through mid-March, the region is influenced alternatively by moist, tropical air masses from the south and from cold, dry continental air masses from the north. The general extratropical storm track is to the north of New Orleans but occasional systems do develop offshore of the city, causing sudden drops in temperature and an increase in precipitation. The cold Mississippi River water and the surrounding marsh areas increase the occurrence of fog in the late winter and early spring months, particularly when light southerly winds are advecting warm, moist tropical air over the area. A rainy period between mid-December through mid-March occurs annually, with the remaining fall/spring months (e.g., October/November, April/May) being relatively dry. The dominant rainfall event during the summer are thunderstorms. Severe thunderstorms with damaging winds are rare. However, the area is subject to the occasional landfalling hurricane. Waterspouts are common in the offshore area, but rarely cause property damage of loss of life (NOAA, 1992l).

D.2.2.15 Oakland, CA

Oakland, located on the eastern shore of the San Francisco Bay, is directly opposite San Francisco. It is the second largest port on the Bay and is a leading containership terminal on the Pacific Coast. The approach to San Francisco and the transit across the Bay to Oakland is open, however, there are restricted areas such as passing under the Golden Gate and Oakland Bay Bridges. There is considerable traffic in the Bay area. A Federal project channel depth of 10.6 m (35 ft) exists to and in the outer harbor. The same depth is maintained in part of the inner harbor. The width passage from the ocean to San Francisco Bay is reduced to approximately 1,125 m (0.7 mi) at the Golden Gate Bridge pier. The distance from the Golden Gate Bridge to the entrance of Oakland Harbor is less than 16 km (10 mi) (DOC, 1992b). A map of the port is shown in Figure D-46.

Oakland is a huge multi-terminal port complex consisting of Outer, Middle, and Inner Harbor cargo terminals leased to terminal operators and/or container shipping lines. The Port of Oakland is part of the Oakland Municipality. The Port Administration is strictly a "landlord" owner and does not operate any facilities. There is a growing trend for "secondary" use by other shipping lines of privately leased terminals, such as Matson's Outer Harbor 7th Street terminal—Berths 32-34, blurring the distinction between public and private use (Jane's, 1992; AAPA, 1993). The port handled over one million 20-ft equivalent units of containerized cargo in 1992 (AAPA, 1994) Public use container and general cargo facilities include:

Outer Harbor: The Seventh Street Marine Container Terminal, Berths 37 and 38, has three container cranes, 14.4 ha (35.6 acres) of terminal area, and storage for over 2,500 20-ft equivalent units. The Outer Harbor Public Container Terminal, Berth 23, has two container cranes, 16.2 ha (40 acres) of terminal area, and storage for over 3,500 20-ft equivalent units. The Bay Bridge Terminal, Berths 8-10, (a combination general cargo (breakbulk), container, and roll-on/roll-off facility) has 20.6 ha (50.9 acres) of terminal area and 7,072 m^2 (76,130 ft^2) of covered storage. This terminal was inoperative for several years due to earthquake damage sustained in 1989 but is now back in operation (Adams and Renteria, 1994). Quay lengths are as follows: Seventh Street Marine Containers Terminal — 592 m (1,942 ft) of marginal wharf; OHPCT — 274 m (900 ft) of marginal wharf; and Bay Bridge Terminal — 926 m (3,038 ft) of marginal wharf.

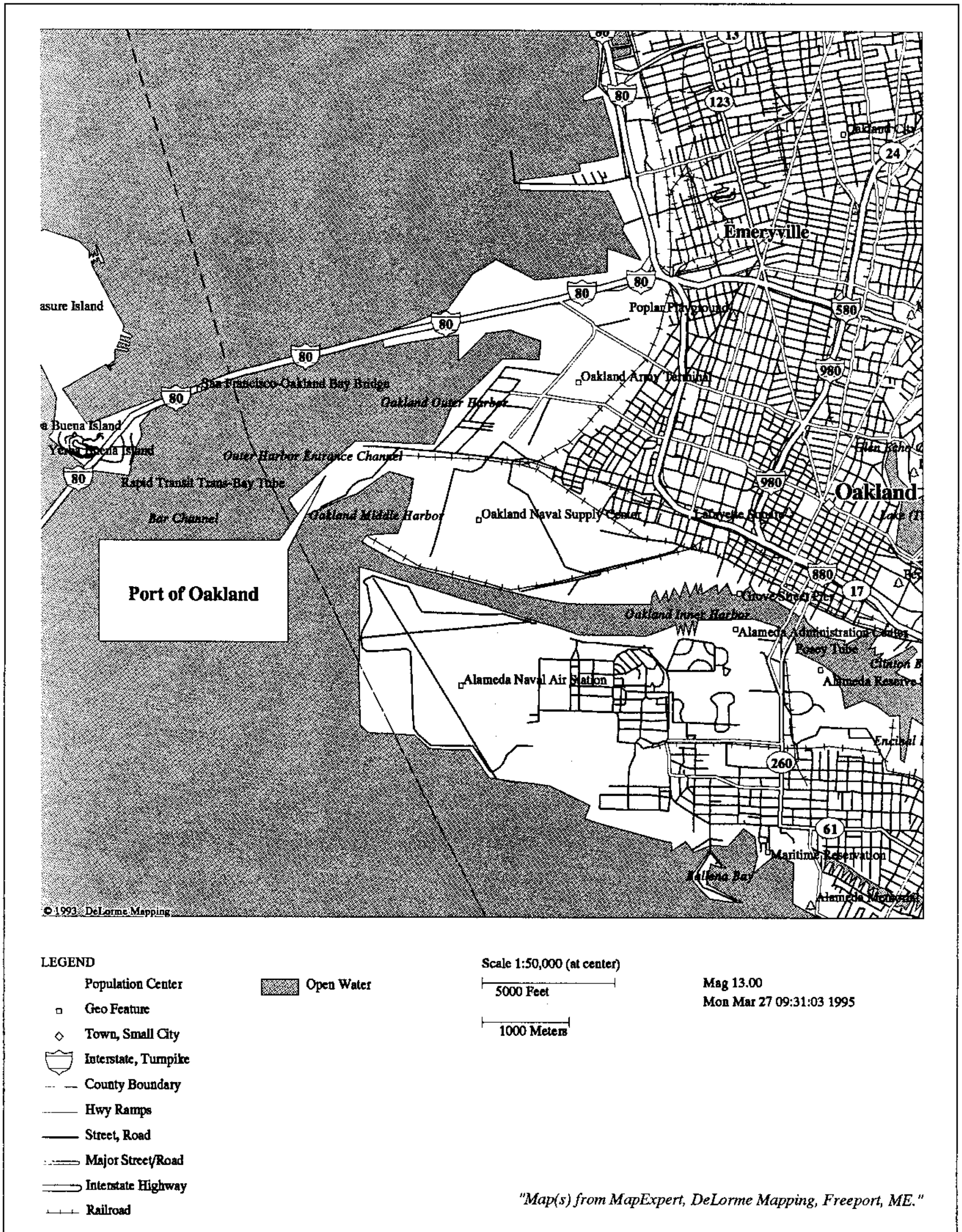


Figure D-46 Map of the Port of Oakland, CA

Crane capacities at Seventh Street Marine Containers Terminal include two 40.6 metric ton (45 ton) container cranes and one 30.5 metric ton (34 ton) container crane. Outer Harbor Public Container Terminal has crane capacity of two 40.6 metric ton (45 ton) container cranes (Jane's, 1992; AAPA, 1993).

Inner Harbor: The Charles Howard Terminal, Berths 67 - 69, has three container cranes, 19.8 ha (48.9 acres) of terminal area, and storage for over 3,000 20-ft equivalent units. Quay lengths at Charles Howard Terminal are as follows: two marginal wharves of 501 m each (1,642 ft) and one 173 m (568 ft) wharf. The Charles Howard Terminal has crane capacity consisting of two 40.6 metric ton (45 ton) container cranes and one 50.5 metric ton (56 ton) container crane (Jane's, 1992; AAPA, 1993).

Seventh Street Marine Containers Terminal, Outer Harbor Public Container Terminal, and Charles Howard Terminal have depths alongside at mean low water of 12.2 m (40 ft). Approach channels are currently limited to 10.6 m (35 ft). A dredging program to 12.8 m (42 ft) is scheduled for completion by 1995 (Jane's, 1992; AAPA, 1993).

Located just south of the Oakland Bay Bridge, the Port of Oakland has immediate access to Highway I-80 for shipments to Idaho National Engineering Laboratory and/or transcontinental shipments, and Highways I-580/I-5 for east coast shipments via the southern route I-40. The truck route from Seventh Street Marine Containers Terminal to the interstate appears to be almost entirely within the port complex in an area dedicated to cargo handling and shipping functions. The Port of Oakland is served by the Union Pacific, Southern Pacific, and Santa Fe Railroads (D&B, 1993). The port has an intermodal container transfer facility, but there does not appear to be direct rail service to container berths at the Seventh Street Marine Containers Terminal (Jane's, 1992; AAPA, 1993).

The Port of Oakland is served by many of the world's largest container lines, including American President Lines, Atlantic Container Lines, Australia-New Zealand Container Line, Cho Yang, DSR Senator Line, EAC Lines, Hanjin Shipping Co., Hapag-Lloyd, Hawaiian Marine, Hyundai, Italian Line, "K" Line, Maersk Lines, Matson Navigation Co., Mitsui OSK, Neptune Orient, NYK Lines, OOCL, Sea-Land Service, and Yang Ming Line (Jane's, 1992; AAPA, 1993; D&B, 1993). Four additional lines switched from San Francisco to Oakland in 1994 (Mitchell, 1994; Adams, 1994).

Other Pertinent Information: Security of the port is provided by perimeter fencing and unarmed guards from the International Longshoremen Union who maintain 24-hour patrol and surveillance (Adams, 1993; Adams and Renteria, 1994). Therefore, it is assumed that foreign research reactor spent nuclear fuel shippers using the port would have to provide their own security force.

The Port Commission has an active ban on the handling of spent nuclear fuel in recognition of community anti-nuclear sentiment which led to a citizens legislative initiative banning such shipments (subsequently struck down by a Federal court). The port handles radioactive and other hazardous materials shipments but officials did not know if Oakland has ever handled spent nuclear fuel shipments (Adams, 1993). The available data shows that Oakland has received spent nuclear fuel shipments, with the last shipment in 1988 (NRC, 1993).

Outer Harbor container and general cargo terminals are situated at the entrance to the port and there appears to be little or no conflict with other hazardous cargoes including petroleum products shipped through the port's breakbulk and liquid bulk terminals located within the Inner Harbor (Adams, 1993; Adams and Renteria, 1994).

Emergency response capability is the responsibility of the individual terminal operators. Each terminal operator must have an Emergency Contingency Plan approved by the Port Commission and the U.S. Coast Guard. The Oakland Fire Department has a hazardous materials response team, and the response time for emergencies is about five minutes (Adams and Renteria, 1994). Beginning in November 1994, the port is increasing its emergency response capabilities. Financed by a new $50 million bond, the port is adding a new fire station, an Emergency Operations Center, new fire boats, a completely equipped hazardous materials van, and a fire-fighting bucket to be lifted in by helicopter. The port also has agreements with neighboring cities (Berkeley, San Leandro, and Alameda) for emergency response (Adams and Renteria, 1994). It is the responsibility of individual terminal organizations and/or the port to provide hazardous materials instruction to the longshoremen (Adams, 1993; Adams and Renteria, 1994).

The Seventh Street Marine Containers Terminal is located in the Outer Harbor terminal complex seaward of the downtown Oakland business district, in an area primarily dedicated to port industrial usage with excellent connections to highways and rail service. However, the port is located in a large urban area in which congestions are to be expected. The San Francisco Bay Area has had only 31 collisions, but 21 fires were reported during the period 1991 to 1993—the worst fire record for major West Coast ports (USCG, 1994b).

There are no known areas of special environmental concern; however, there is strong concern for preservation of the environment, and this area is prone to severe earthquakes. On April 18, 1906, the Bay area was subjected to one of the largest recorded earthquakes in the contiguous United States, a Modified Mercalli Intensity XI (Bolt, 1978), due to movement along the fault line separating the Pacific and Continental tectonic plates (Hamilton, 1976). The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Oakland, the Uniform Building Code requires buildings to withstand wind speeds up to 110 km/hr (70 mph). Since the port is located in a very high seismic zone (the highest Uniform Building Code ranking), buildings must be constructed to withstand an acceleration of 0.40 g.

The 1990 population within 16 km (10 mi) of the preferred port terminals was 1,387,611. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,080,000; Oak Ridge Reservation, 786,000; Idaho National Engineering Laboratory, 367,000; Hanford Site, 359,000; and Nevada Test Site, 482,000. Populations along rail routes to these sites are slightly larger for Savannah River Site, Hanford Site and Nevada Test Site, but slightly smaller for Oak Ridge Reservation and Idaho National Engineering Laboratory. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,490 km (2,791 mi); Oak Ridge Reservation, 4,120 km (2,563 mi); Idaho National Engineering Laboratory, 1,550 km (963 mi); Hanford Site, 1,410 km (875 mi); and Nevada Test Site, 1,160 km (719 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The area around the terminal includes the San Francisco Bay to the south and the San Pablo Bay to the north. The Farallon Islands Game Refuge and the Point Reyes-Farallon Islands National Marine Sanctuary lie along the passageway to San Francisco. The San Francisco Bay National Wildlife Refuge and the Redwood Shores Ecological Reserve, both within the San Francisco Bay, are located 33 and 30 km south (20 and 18 mi), respectively from the Port of Oakland. The San Pablo Bay National Wildlife Refuge is located approximately 160 km (99 mi) north of the port.

San Francisco Bay

The Central Bay portion of the San Francisco Bay has several usage classifications, including industrial process supply, industrial service supply, navigation, water contact recreation, noncontact water recreation, ocean commercial and sport fishing, wildlife habitat, preservation of rare and endangered species, fish migration, fish spawning, shellfish harvesting, and estuarine habitat (State of California, 1986).

The San Francisco Bay comprises the largest estuarine ecosystem in California. The estuary encompasses a range of aquatic habitats, from the fresh and brackish waters of the Sacramento-San Joaquin River Delta to the saline waters of the Central and South Bay. The estuary provides habitat for a variety of aquatic species, some of which are important to commercial and recreational fisheries. These waters serve as a nursery area for marine, anadromous, and estuarine species, and provide a migration corridor for several anadromous species. Striped bass, Chinook salmon, steelhead trout, sturgeon, American shad, and English sole support important recreational fisheries in the estuary. Popular recreational fisheries in the Delta also include white catfish, largemouth bass, and sunfish (U.S. Army, 1994). In addition, the area around the port has populations of the common littlenecked clam, the soft-shelled clam, striped bass and flatfish, the California clapper rail, and the salt-marsh harvest mouse (FWS, 1981b).

Historically, marshlands bordering the Bay covered some 300 mi^2; diking for agriculture and filling for development has reduced the marshlands to about 75 mi^2 (U.S. Army, 1994). The marshes and mudflats remaining along the margins of the Bay are very productive and provide habitat for a large number of birds and other wildlife. For instance, the area around the port has populations of the California clapper rail and the salt-marsh harvest mouse (FWS, 1981b). The Bay is a key resting, feeding, and wintering area for birds on the Pacific Flyway. This area annually supports a large number of shorebirds, wintering waterfowl, raptors, seabirds, and passerines. Shorebirds, wading birds, waterfowl, seabirds, songbirds, and other species migrate through the entire coastal zone in the San Francisco area (FWS, 1981b).

Several threatened or endangered species are known to occur or have the potential to occur occasionally or periodically in the San Francisco Bay area. These species include the California least tern, California brown pelican, the American peregrine falcon, and the winter-run chinook salmon (U.S. Army, 1994). The least terns breed in California from mid-May to August and nesting colonies are located on open flat beaches, sand flats, and bare dirt areas with sparse vegetation. The least tern generally migrates from the Bay Area in August and winters south of the United States. The California brown pelican uses the open waters of the central San Francisco Bay for feeding; they roost on rocks, jetties, and piers in the area. Although no brown pelicans breed in the San Francisco area (Bay or offshore), thousands move north and roost on coastal rocks during the June through October nonbreeding season. Several thousand pelicans summer in the San Francisco area. The American peregrine falcon is considered rare in the region. It formerly bred on the Farallon Islands, and though it has yet to breed there again, winter residents have returned and have stabilized in number. The American peregrine falcon is most common to the San Francisco Bay area during the winter, when migrants from farther north concentrate in the estuary. The nesting season is from spring thorough early summer, and several pairs nested on the San Francisco-Oakland Bay Bridge (U.S. Army, 1994). California condors and bald eagles are also found in the coastal zone around San Francisco Bay (FWS, 1981b). The winter-run chinook salmon passes through the Sacramento-San Joaquin Delta, San Pablo Bay, and San Francisco Bay during their upstream and downstream migrations. The adults are present in the Bay area from November to May, and the smelts migrate through the Bay from November through May. The winter-run chinook is fished commercially in North America from Kotzebue Sound, Alaska, to Santa Barbara, California (U.S. Army, 1994).

Open Ocean

Several threatened or endangered species occur either occasionally or periodically in the ocean offshore of the San Francisco area. These include the humpback whale, the blue whale, the sperm whale, and the Stellar sea lion (U.S. Army, 1994). The humpback whale, which has a worldwide range, is typically found in the San Francisco area from March through January. Summer feeding occurs from the Aleutian Islands to the Farallon Islands. The greatest number of blue whales within the Farallon Basin occurs in summer and early fall. The sperm whale regularly occurs in the Gulf of the Farallones in deep oceanic waters, and is rarely reported over the shelf. The Stellar sea lion ranges from California to the Bering Sea. Stellar sea lions have rookeries on Southeast Farallon Islands (as well as other California and Pacific coast sites). The sea lion breeds in the late spring and summer.

Climatic Conditions

The Oakland, CA, area is classified as a marine climate, which is characterized by mild and moderately wet winters, with cool, dry summers. The winter rains, which occur between November and March, account for over 80 percent of the total annual precipitation. Additionally, severe winter storms, with gale-force winds and heavy rains do occur occasionally. The diurnal temperature range is moderated substantially by marine environment. The summer weather is dominated by a cool sea breeze circulation and a sea fog that arrives in the late evening over the area. The fog generally burns-off in the early morning hours, resulting in relatively sunny summer days (NOAA, 1993d).

D.2.2.16 Palm Beach, FL

The Port of Palm Beach is located 2.0 km (1.1 mi) west of the entrance to Lake North Worth Inlet, which consists of a dredged cut, protected by two jetties, through the barrier beach which forms the resort city of Palm Beach. The port borders the communities of Riviera Beach on the north and West Palm Beach on the south, the latter being connected to Palm Beach by highway bridges spanning Lake Worth. The Port of Palm Beach is 110 km (68 mi) north of Miami and 417 km (259 mi) south of Jacksonville. A Federal project provides for a 10.7 m (35 ft) deep entrance channel with a 10.1 m (33 ft) inner channel to a turning basin of the same depth. The 121.9 m (400 ft) wide entrance channel narrows to 91 m (300 ft) and leads into a 442 m by 399 m (1,450 ft by 1,310 ft) turning basin. Port Authority-owned Peanut Island is located between the inlet entrance and Port of Palm Beach terminals. According to the port's 1993 Annual Report, the controlling depth of the entrance and turning basin was 10.1 m (33 ft) to not less than 7.6 m mean low water (25 ft) at the northern terminal extension (POPB, 1994). A map of the port is shown in Figure D-47.

The Port of Palm Beach is a landlord port with 77 ha (190 acres) of land. The Terminal has two slips and four marginal wharves totalling 1,536 m (5,039 ft) of berthing, including six roll-on/roll-off ramps. Pilots limit the maximum size of ships entering the port to 192 m (630 ft) in length. Total tonnage for the fiscal year ending September 1993 was 3,694,034 metric tons (4,071,934 tons), including 158,172 20-ft equivalent units [(1,005,190 metric tons)(1,108021 tons)] of containerized cargo. The port owns one 228 metric ton (251 km) crawler crane, but containers are either handled by ship's gear or with local stevedoring equipment. Primary commodities handled are containerized general cargo, sugar, molasses, and fuel oil for two local power plants (POPB, 1994; AAPA, 1994; Mets, 1994).

Port of Palm Beach Berths 5-6, 7-11, and Berths 12-17 are operated as public terminals for container handling, general cargo, roll-on/roll-off, cruise lines, and heavy lift cargoes. Berths 5 and 6 have 10.1 m (33 ft) depth alongside and have rail service on the pier, which is owned and operated by the port

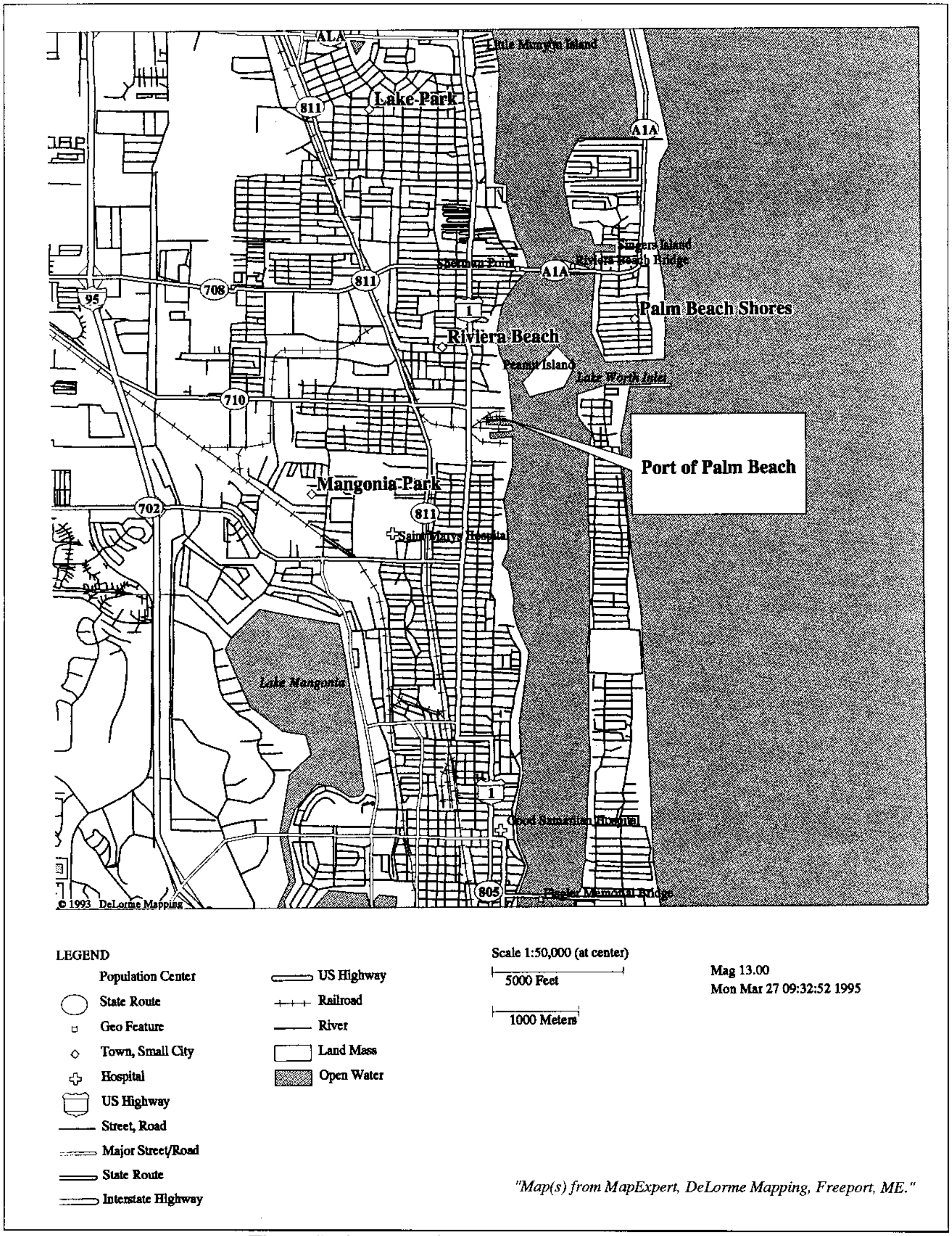


Figure D-47 Map of the Port of Palm Beach, FL

connecting with the Florida East Coast Railroad. The Port of Palm Beach is about 2.4 km (1.5 mi) from I-95 and 8.1 km (5 mi) from the entrance to the Florida Turnpike. The route is through light commercial and residential areas (AAPA, 1994).

Other Pertinent Port Information: Palm Beach has an around-the-clock watchman service, is fenced and lighted and has only one controlled entrance/exit. The port handles explosives and other hazardous goods and according to a port official, the port does not have a prohibition against handling spent nuclear fuel (Mets, 1994). As with other small, multi-use ports, there is some apparent conflict between the handling of petroleum products, cruise ship passengers, and spent nuclear fuel all within the confines of a relatively small, environmentally sensitive harbor complex.

The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Palm Beach, the Uniform Building Code requires buildings to withstand wind speeds up to 160 km/hr (100 mph). The port is located in a very low seismic zone with an acceleration of less than 0.075 g.

Negotiations for the sale of the port's Peanut Island, mentioned above, to the Florida Inland Navigation District are currently underway. Use of the island would be permanently limited to a partial dredge spoil area, as well as habitat preservation, and a passive recreation area (POPB, 1994; Mets, 1994).

The 1990 population of the combined port area (Riviera, Palm, and West Palm Beach) was approximately 115,000, and the average are density was about 650 persons/km^2 (1,600 persons/mi^2). The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are approximately (based on data for nearby Port Everglades): Savannah River Site, 240,000; Oak Ridge Reservation, 350,000; Idaho National Engineering Laboratory, 780,000; Hanford Site, 790,000; and Nevada Test Site, 800,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes for nearby Port Everglades are approximately: Savannah River Site, 1,125 km (700 mi); Oak Ridge Reservation, 1,366 km (850 mi); Idaho National Engineering Laboratory, 4,501 km (2,800 mi); Hanford Site, 5,145 km (3,200 mi); and Nevada Test Site, 4,662 km (2,900 mi). Distances along rail routes are slightly longer.

Climatic and environmental conditions are similar to those reported for Port Everglades in Section D.2.2.18.

D.2.2.17 Philadelphia, PA

Philadelphia, one of the chief ports of the United States, is located at the junction of the Delaware and Schuylkill Rivers, approximately 130 km (81 mi) above the entrance to the Delaware Capes. Access to the port is via the Delaware River through the Delaware Bay. Situated directly across the Delaware River from Philadelphia is Camden, NJ, an important shipping center. The shipping activities of the two cities are closely allied; large quantities of general cargo are handled at the Philadelphia port in both domestic and foreign trade. Access to the port is gained via the Delaware Bay and Delaware River (DOC, 1993c). A map of the port (including Camden, NJ immediately opposite) is shown in Figure D-48.

The Delaware Bay has natural depths of 15.4 m (50 ft) or more for a distance of 8 km (5 mi) from the entrance. A Federal project provides depths of 12.2 m (40 ft) from the sea through the Delaware Bay and River to Philadelphia. There are restrictions on the passage through the Delaware Bay and up the Delaware River, such as a traffic separation scheme established off the entrance to the Delaware Bay.

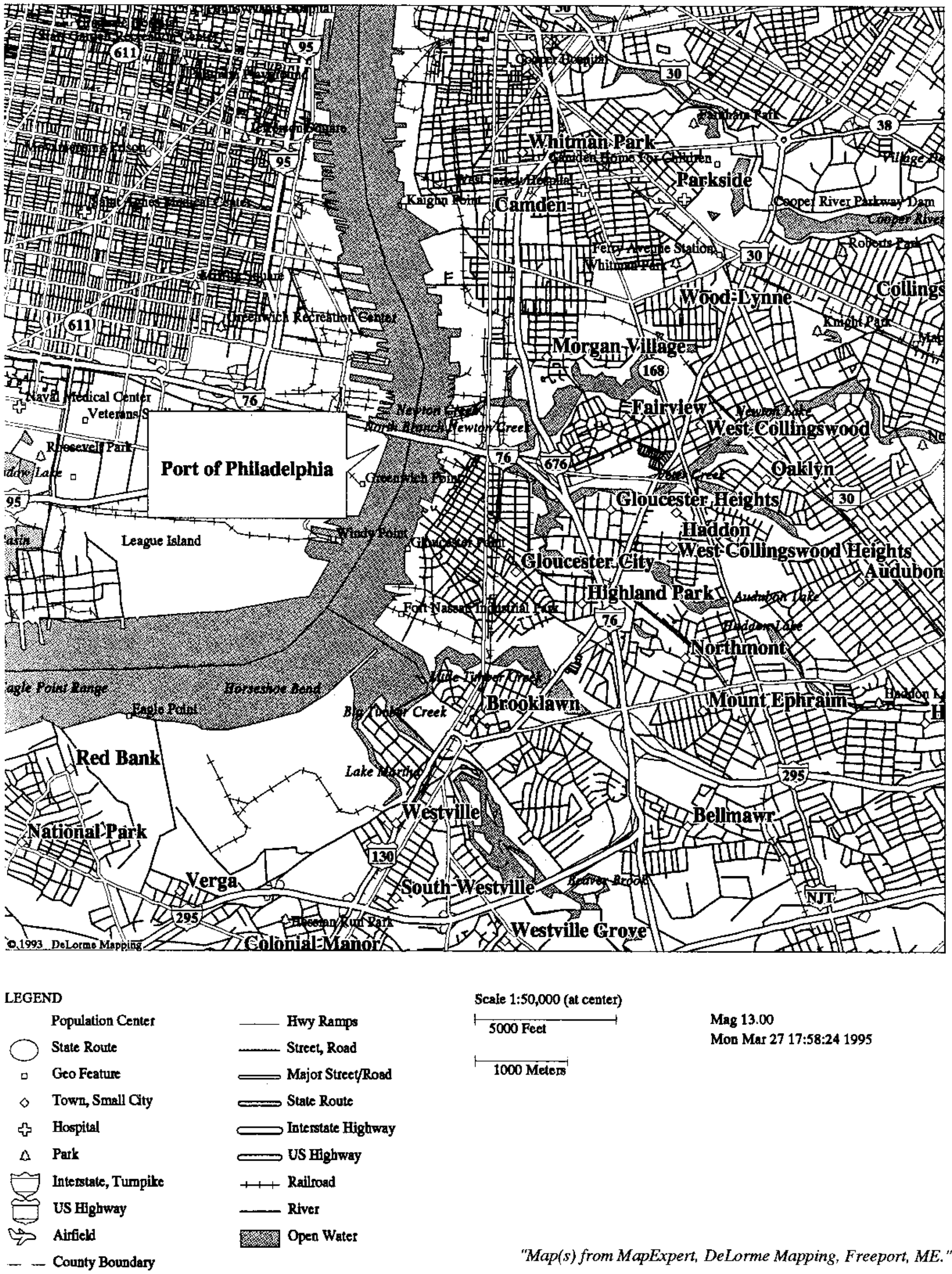


Figure D-48 Map of the Ports of Philadelphia, PA, and Camden, NJ

Ships going to Philadelphia must also pass under the Delaware Memorial Bridge. Roughly 90 percent of the 58,831,000 metric tons (64,849,000 tons) of cargo handled in 1991 were bulk cargo, and a large share of that is known to be crude oil and refinery products (DOC, 1993c; AAPA, 1993).

With the exception of some privately owned terminals, general cargo facilities (including container terminals on the west side of the Delaware River) are owned by the City of Philadelphia. Across the river on the New Jersey side, cargo terminals belong to the South Jersey Port Corporation, a state agency that operates two terminals and leases the remaining ones to private companies. The Philadelphia Regional Port Authority, apparent successor to the Philadelphia Port Corporation, is responsible for City-owned terminals leased to private companies under a landlord-type operation. The Delaware River Port Authority functions as a port planning and economic development division for the facilities controlled by the Philadelphia Regional Port Authority and the South Jersey Port Corporation.

The South Jersey Port Corporation operates two multi-berth terminals, Beckett Street and Broadway. These terminals primarily handle breakbulk general cargoes. Both breakbulk and containers (approximately 4,000 to 5,000 per year) are handled at the Beckett Street terminal during the winter (Castagnola, 1994). Beckett Street terminal has two container cranes, one with a 36.6 metric ton (40 ton) capacity and one with a 77.1 metric ton (85 ton) capacity (Castagnola, 1994). Pier 6, one of the Broadway Berths, is leased to a private company and is equipped with one 72.6 metric ton (80 ton) capacity multi-purpose container crane and one 40.6 metric ton (45 ton) container crane (Jane's, 1992; AAPA, 1993).

Principal container handling facilities owned by Philadelphia Regional Port Authority are the Packer Avenue Terminal, a combination breakbulk/container terminal, and Tioga Container Terminals. The former is located immediately downstream of the Walt Whitman Bridge at the south end of the City's waterfront. The Tioga Terminal is approximately 9.7 km (6 mi) further upstream. Both the Packer Avenue and Tioga Terminals have a depth alongside at mean low water of 12.2 m (40 ft) (Jane's, 1992; AAPA, 1993).

Packer Avenue Terminal: The Packer Terminal is equipped with two 45 metric ton (50 ton) container cranes and one 37.5 metric ton (41 ton) container crane, and has a paved open storage area of 23.5 ha (77 acres). The terminal has 1,184 m (3,885 ft) of marginal wharf (Jane's, 1992; AAPA, 1993).

Tioga Terminal: Tioga has 20.2 ha (50 acres) of paved open storage and is equipped with two 45 metric ton (50 ton) container cranes. The terminal has 796 m (2,612 ft) of marginal container berth; and a 185 m (610 ft) roll-on/roll-off berth.

Both the Packer Avenue and Tioga Terminals are relatively short distances from I-95, which parallels the River and is estimated to be within 0.8 km (0.5 mi) of the container terminals. The Packer Avenue Terminal is served by CSX and Conrail; CSX maintains an intermodal terminal just outside the terminal. On the other hand, the Tioga Terminal has Conrail intermodal service at the terminal. However, neither terminal has ship-side trackage (Jane's, 1992; AAPA, 1993).

A partial list of the diverse liner shipping companies serving these Delaware River terminals include: ABC Container Line, PACE, Atlantik Express Lines, Baltic Shipping Lines, Bangladesh Lines, Barber West Africa, Bottachi, Chilean Line, Colombus, Egyptian National Line, Ellerman, ELMA, Empremar Line, Euro Line, Frota Amazonica, Grandcolumbiana, Hapag-Lloyd, Hoegh, Hyundai, Independent Container Line, Maersk, Netumar, Pakistan National, Shipping Corp of India, SITRAM, Tokai, and Toko (Jane's, 1992).

Other Pertinent Information: The container terminals are fenced with controlled access and 24-hour security. It is not known what arrangements exist for temporary storage of hazardous materials, but it is likely such storage is available in a large port facility (Castagnola, 1993). Spokespersons for the South Jersey Port Corporation (Castagnola, 1993; formerly with the Philadelphia Regional Port Authority) and the Philadelphia Regional Port Authority (Menta, 1993) were unaware of any restrictions on handling spent nuclear fuel, but indicated this was outside their areas of expertise. There are several major oil refineries along the Delaware River below and west of the City of Philadelphia on the Schuylkill River. However, there does not appear to be any serious conflicts in close proximity to the Packer Avenue or Tioga Terminals.

The South Jersey Port Corporation relies on state hazardous materials agencies and the Camden Fire Department for emergency response to the terminals it operates. Private operators are responsible for their own terminals, but basically rely on the fire department. Being a landlord port operator, it is assumed Philadelphia Regional Port Authority terminal operators rely on the Philadelphia Fire Department to respond to hazardous materials incidents. It is not known if there is any hazardous materials training by the Philadelphia Regional Port Authority or the South Jersey Port Corporation (Castagnola, 1993). There are no known special environmental concerns. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Philadelphia, the Uniform Building Code requires buildings to withstand wind speeds up to 120 km/hr (75 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The Philadelphia and Camden waterfronts have become tourist centers due to historical sites and a new aquarium on the Camden waterfront. Veterans Stadium and the Spectrum are located in relatively close proximity to Packer Avenue, as are the Philadelphia Navy Yard and the Philadelphia International Airport.

All terminals in the Greater Philadelphia area are basically located in densely developed and populated industrial/commercial areas. The 1990 population within 16 km (10 mi) of the port terminals (including Camden, NJ) was 1,915,775. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 397,000; Oak Ridge Reservation, 335,000; Idaho National Engineering Laboratory, 513,000; Hanford Site, 622,000; and Nevada Test Site, 756,000. Populations along rail routes to these sites are much larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,190 km (741 mi); Oak Ridge Reservation, 1,090 km (680 mi); Idaho National Engineering Laboratory, 3,950 km (2,452 mi); Hanford Site, 4,610 km (2,868 mi); and Nevada Test Site, 4,220 km (2,623 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The Delaware River at Philadelphia is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms typically found in the waters of this area include: American shad, Atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel (FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. The U.S. Fish and Wildlife Service reported that except for occasional transient species, no federally listed or proposed threatened or endangered species are known to exist in the port's impact area (Perry, 1994).

The Port of Philadelphia is located within Zone 3 (tidal river) of the Delaware River. Protected water used for Zone 3, which encompasses River Mile (RM) 95-108.4, are water supply (agricultural, industry, and public), wildlife, resident fish maintenance, anadromous fish passage, secondary contact, and navigation (DRBC, 1994). However, several uses within Zone 3 are currently impacted, including: 1) fish and other aquatic life due to low dissolved oxygen levels from point source discharges; and 2) fish and shellfish consumption due to chlordane and polychlorinated biphenyl contamination from point and nonpoint source discharges.

Climatic Conditions

The climate of Philadelphia is moderated by the Appalachian Mountains to the west and the Atlantic Ocean to the east. These geographic features cause periods of extreme temperatures to be short-lived in this region (generally, four days). On occasion during the summer months, the area is dominated by maritime tropical air masses, which contribute to elevated local temperature and humidity levels. The average annual precipitation (41.42 in) is relatively evenly distributed throughout the year, with maximum amounts occurring during the late summer months. The summer precipitation regime is dominated by localized thunderstorms and is subject to the influence of the urban heat island effect and local topography, which create varying rainfall amounts across the city for an individual event. Singular snowfall events that generate accumulated totals of greater than 10-in have a 5-year recurrence interval on average. The prevailing wind direction has a bimodal distribution, being southwesterly during summer and northwesterly in the winter months. The annualized average prevailing wind direction is from the west-southwest. Due to Philadelphia's inland location, destructive winds are comparatively rare from tropical cyclones and tornadoes. High winds are generally associated with frontal passages/low pressure systems in winter and thunderstorms in summer months. However, flooding on the Schuylkill River normally occurs twice annually usually associated with strong thunderstorms, with the duration of these events generally lasting less than 12 hrs. The Delaware River is rarely observed at or above flood stage (NOAA, 1992h).

D.2.2.18 Port Everglades, FL

Port Everglades is a major deepwater port located on Florida's southeast coast. It is located immediately off the Atlantic Ocean along the Inland Waterway, within the three cities of Hollywood, Fort Lauderdale, and Dania (DOC, 1993d; D&B, 1993; Southern Shipper, 1993). The major port facilities are immediately inside the harbor entrance, approximately 1.6 km (1 mi) from the south jetty. The approach to Port Everglades is open, and a relatively short 140 m (450 ft)-wide channel leads directly from the Atlantic Ocean to the port facilities. A Federal project provides for depths of 12.8 m (42 ft) to the main port facilities (DOC, 1993d; D&B, 1993; Jane's, 1992; Southern Shipper, 1993; AAPA, 1994; PEA, 1993).

Port Everglades consists of 850 ha (2,100 acres) of land, of which 360 ha (890 acres) are owned by the Port Everglades Authority Commission (Port Everglades Authority). Considerable foreign commerce passes through Port Everglades, in addition to substantial passenger traffic. Many of the world's large passenger vessels call at Port Everglades (it claims to be the world's second-busiest cruise port) (Southern Shipper, 1993). It is a multi-terminal port with more than 3,800 ship calls annually. The port handles over 2,000,000 cruise passengers as well as 17 million tons of cargo, including 1.4 million metric tons (1,600,000 tons) of containerized general cargo (over 100,000 20-ft equivalent units) and over 14 million metric tons (16,000,000 tons) of bulk cargoes (dry/liquid/scrap) in 1991 (FS, 1992). Port Everglades is also a liberty port for the U.S. and North Atlantic Treaty Organization Navies, and is host to facilities operated by the U. S. Naval Surface Warfare Center in Fort Lauderdale. The port is also bordered on the east by a large State Park and seashore recreation area. A map of the port is shown in Figure D-49.

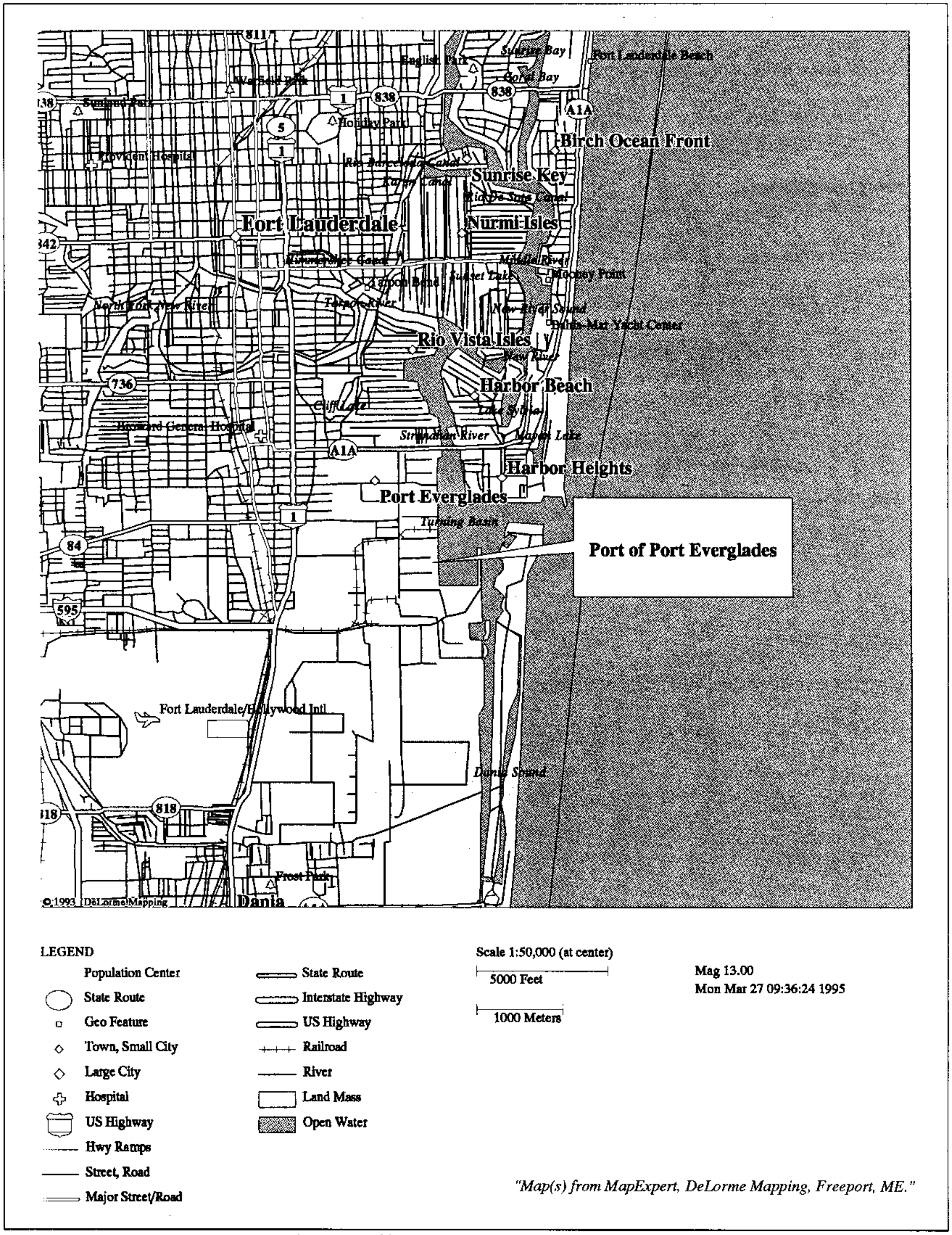


Figure D-49 Map of Port Everglades, FL

The Port Everglades Authority is empowered by the State Legislature to act as the governing entity for the operations, maintenance, and management of port and harbor facilities located within the port's jurisdictional area. The Authority is the governing body responsible for strategic planning and policy setting. In November 1994, governing responsibility for the seaport was transferred to the Broward County Government (PEA, 1993).

Principal container handling facilities at the port include Midport and Southport (there is also a Northport terminal as well).

Midport: Midport has three berths (16, 17, and 18), with three container cranes and 17.8 ha (44 acres) of open storage, and is located just inside the harbor entrance. The berths have a total of 502 m (1,650 ft) of marginal wharf area and depths of 11.6 m (38 ft) alongside at mean low water. Crane capacities at Midport consist of one 50.8 metric ton (56 ton) container crane and two 30.5 metric ton (34 ton) container cranes. Access to Midport is via Port Road, Highway 84, which intersects with U.S. 1, I-95, and all major interstate highways. Highway access appears to be exclusively through port property and/or adjacent industrial-use land. Port-owned trackage is leased to the Florida East Coast Railroad, which maintains an intermodal container yard 1.6 km (1 mi) outside the port on State Road 84. There is trackage to Berth 4, but there does not appear to be any rail lines presently serving the container terminals. There are two roll-on/roll-off ramps at Midport (Jane's, 1992; D&B, 1993; Southern Shipper, 1993; AAPA, 1993 and 1994; PEA, 1993).

Southport: Southport is a new 62.7 ha (155 acre) container/roll-on/roll-off complex at the southern end of Port Everglades Harbor. Southport consists of Berths 31 and 32 with a combined 610 m (2,000 ft) of marginal wharf. Berths 30-33C at Southport have depths of 13.4 m (44 ft) alongside at mean low water. Crane capacities for the berths at Southport include three 40.6 metric ton (45 ton) Post-Panamax container cranes. Access to Southport is via the new Port Everglades Expressway (I-595), which begins just outside the Terminal and connects with I-95, the Florida Turnpike, I-75, and State Highway 84, the cross-Florida Everglades Expressway. Highway access appears to be exclusively through port property and/or adjacent industrial-use land. Southport has a 25.5 ha (63 acre) container yard with storage for up to 5,100 containers on chassis or up to 7,872 grounded and stacked 20-ft equivalent units. There are three roll-on/roll-off ramps at Southport, and an additional three ramps are located at Northport.

Port Everglades is served by over a dozen container and breakbulk Liner Shipping Companies offering sailings to major ports of the world, including South and Central America, Caribbean Islands, North Europe, the United Kingdom, Scandinavia, Spain, the Mediterranean, and the Mid-East. Ship lines include Arawak Caribbean Line, Atlantic Cargo Service, Crowley American Transport, Nedlloyd, Inc., P & O Containers, Ltd., Sea-Land Service, Inc., Tecmarine Lines, and Orient Overseas Container Line.

Other Pertinent Information: Twenty-four-hour security is provided by the Broward Sheriff's Office (BSO). All terminal and container facilities are secured with fencing and controlled gates. The port also has its own 65-member Public Safety Department (fire department), fire fighting equipment and vehicles, and a fireboat for first response to hazardous materials incidents. It is backed up by fire departments from Fort Lauderdale and Hollywood, and Broward County's hazardous materials team. Terminal operators are responsible for their own emergency response arrangements. The Public Safety Department basically functions as an emergency coordinating group. Hazardous materials training is carried out by the municipal agencies responsible for emergency response within the port area. It is not known if the Port Authority itself conducts any training. There appears to be ample space at Southport for temporary storage of hazardous cargoes.

There are no known restrictions to the handling of spent nuclear fuel. However, Item 240 of the Port Everglades Authority Tariff states that explosives, hazardous, or highly flammable commodities or materials may only be handled through special arrangement with the Port Authority. Port Officials indicate that their safety policies, which ban oxidizers such as ammonium nitrate and Class A explosives, would also preclude shipments of spent nuclear fuel. As far as is known, spent nuclear fuel shipments have not been handled by the port (Flint et al., 1993).

Port Everglades is the second-largest petroleum distribution facility in the United States (Southern Shipper, 1993). Major oil companies have more than 86 million barrels of tank space for refined petroleum products inshore of the Midport and Northport terminals. With the possible exception of terminal facilities at Southport, which are remote from the tank farms and other conflicting port users, the potential for conflict between cruise ship operations, tanker traffic, and containerized spent nuclear fuel shipments appears to be great.

Southport Terminal is the preferred terminal, as it is relatively remote from the City of Fort Lauderdale, has direct connection to the Interstate Highway system, and is located in a nonresidential port industrial district. The physical layout and constraints of port waterways, however, combined with its intense use by potentially conflicting types of transport (i.e., cruise ships, tankers and tank barges, container and breakbulk vessels, and recreational traffic, plus a State seashore park on its eastern boundary) detracts from its otherwise superb facilities.

The port is subject to severe hurricanes and tropical storms. The likelihood of severe natural phenomena such as high winds and earthquakes is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For Port of Everglades, the Uniform Building Code requires buildings to withstand wind speeds up to 150 km/hr (95 mph). Port Everglades is located in a very low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 714,176. The affected populations within 0.8 km (0.5 m) of the interstate routes to the 5 potential DOE management sites are: Savannah River Site, 244,000; Oak Ridge Reservation, 352,000; Idaho National Engineering Laboratory, 754,000; Hanford Site, 803,000; and Nevada Test Site, 817,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,170 km (726 mi); Oak Ridge Reservation, 1,420 km (884 mi); Idaho National Engineering Laboratory, 4,540 km (2,820 mi); Hanford Site, 5,210 km (3,236 mi); and Nevada Test Site, 4,700 km (2,923 mi). Distances along rail routes are slightly longer.

Environmental Conditions

The State of Florida has classified the waters surrounding Port Everglades as Class III. This classification indicates that the waters are suitable for recreation, and propagation and maintenance of a healthy, well balanced population of fish and wildlife (FL DEP, 1994). As previously noted, Outstanding Florida Waters are generally waters located within national parks, state parks, national seashores, marine sanctuaries, or aquatic preserves. Waters located near Port Everglades designated as Outstanding Florida Waters include the waters within the John U. Lloyd Beach State Recreation Area, West Lake, Snake Warrior Island, North Beach, and the Hugh Taylor Birch State Recreation Area (FL DEP, 1994). These waterways are afforded special protection by the State environmental regulations.

The waters surrounding Port Everglades are characterized as high salinity estuarine habitats (generally greater than 20 parts per thousand). Aquatic species of interest in the vicinity of Port Everglades include: crabs, shrimp, lobster, seatrout, croaker, tarpon, sheepshead, spot, kingfish, drum, silver perch, bluefish, mullet, pompano, pinfish, pigfish, Crevalle jack, grunt, ladyfish, permit, grouper, snapper, jewfish, snook, striped mojarra, and Atlantic bottlenose dolphin (FWS, 1982b).

The John U. Lloyd Beach State Recreation Area is located approximately 4 km (2.5 mi) south of Port Everglades. Additional special land use areas located near Port Everglades are Everglades Wildlife Management Area and Hugh Taylor Birch State Recreation Area. Protected animal species in the Port Everglades vicinity include: the West Indian manatee, loggerhead sea turtle, green sea turtle, and the least tern. Protected plant species in the area are the coontis, sea lavender, and the silver palm (FWS, 1982b).

Port Everglades has several ongoing environmental programs, including the creation of a Manatee Refuge and "Nursery" area within the confines of the port, a wetlands program, and a 22.3 ha (55 acre) permanent mangrove forest and manatee reserve deeded to the Florida Department of Environmental Regulation. The port was awarded the 1991 National American Association of Port Authorities Environmental Award of Excellence (PEA, 1993). The U.S. Fish and Wildlife Service reported that the Port Everglades Midport Terminal is located in designated critical habitat for the manatee (Johnson, 1995).

Climatic Conditions

The climate of this region is essentially subtropical marine, featuring long, warm summers with abundant rainfall, generally followed by a mild, dry winter. The influence of the ocean is seen in the small diurnal temperature range (generally less than 10 degrees) and the rapid warming of any cold air masses that invade this portion of the State. The predominant windflow is from the east-southeast, which generates conditions right at the coast that are often different than those encountered further inland, due to land-induced frictional effects. Hurricanes occasionally affect the area, with the months of September and October exhibiting the highest frequencies. However, destructive tornadoes (not associated with tropical systems) are rare. Waterspouts are frequently spotted offshore during the summer months, but rarely cause any loss of life or property damage (NOAA, 1993b).

D.2.2.19 Richmond, VA

The Port of Richmond Terminal is located on the left ascending bank of the James River, approximately 140 km (89 mi) above the City of Newport News. The port is owned by the City of Richmond and operated by Mecham Overseas Terminal, Ltd. A map of the port is provided in Figure D-50.

Drafts of vessels using the river above Newport News generally do not exceed 4.5 m (15 ft). Vessels drawing more than 7.5 m (24 ft) do navigate it occasionally, but the Virginia Pilots Association restricts ship drafts to 6.7 m (22 ft). A Federal Project provides for dredging depths of 7.6 m (25 ft) to the Richmond Terminals. Numerous stakes, pilings, wrecks and other obstructions are on both sides of the main channel. Travel on the upper river is restricted to daytime hours for ships more than 77.7 m (255 ft) in length (DOC, 1993c). For FY 1993-94, the Terminal handled a total of 445,700 metric tons (491,300 tons) including 313,540 metric tons (345,620 tons) of containerized cargoes (41,286 20-ft equivalent units), and 114,890 metric tons of breakbulk freight (AAPA, 1994; PORT, 1994). Major shipping lines connect the port with the Mediterranean, North Europe, South America, the Middle East, and India (PORT, 1994).

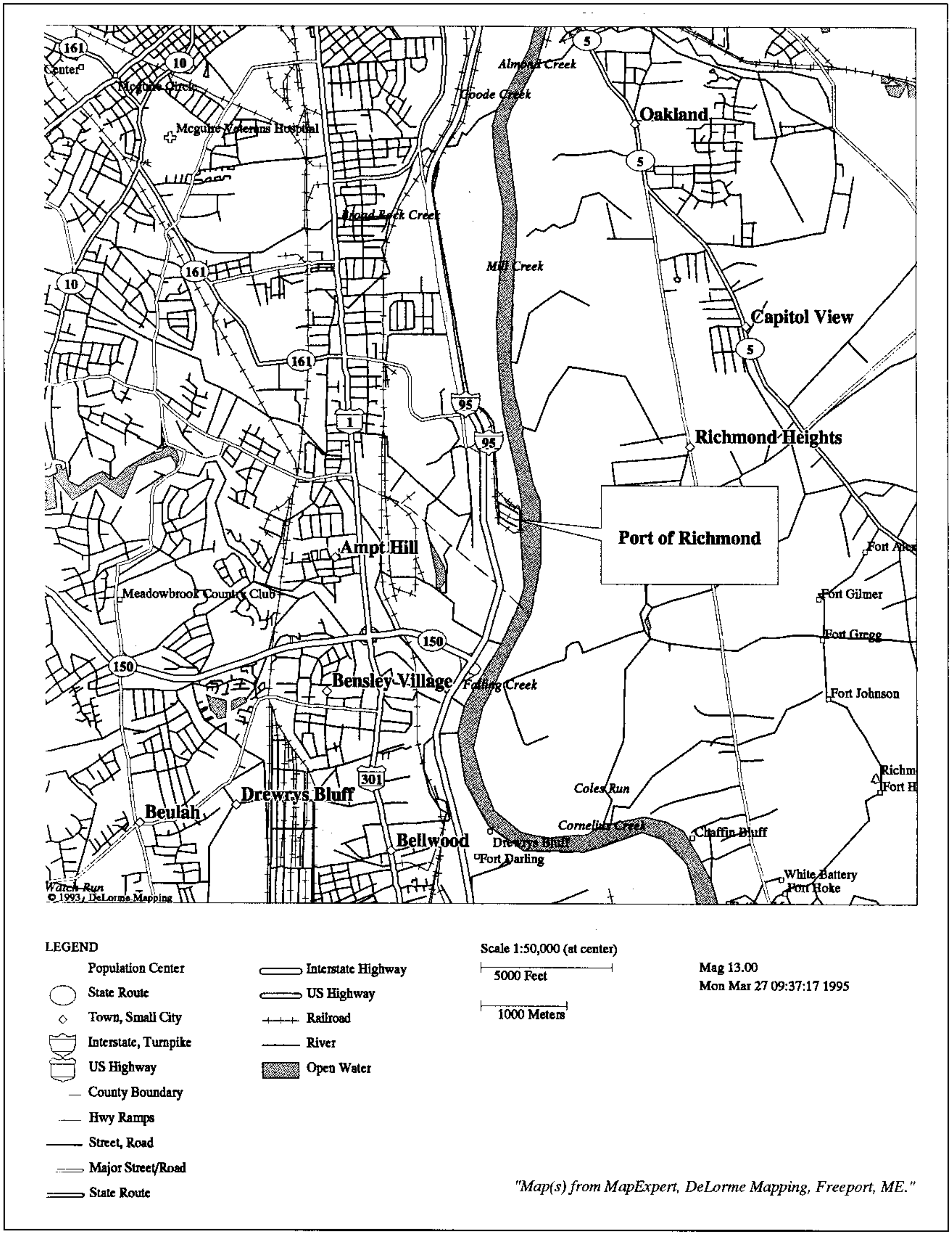


Figure D-50 Map of the Port of Richmond, VA

Richmond Terminal has two marginal berths with a total length of 381 m (1,250 ft) and 7.6 m (25 ft) of water alongside at mean low water. CSX Railroad tracks with multiple sidings serve the port's two warehouses and container storage yards. The Terminal is a container, general cargo, and breakbulk handling facility with roll-on/roll-off vessel and container and trailer on flatcar capabilities. The port has two 209 metric ton (230 ton) and one 319 metric ton (350 ton) capacity crawler cranes outfitted with 22.9 m (75 ft) booms. A new 273 metric ton (300 ton) crane was purchased in April 1994 and set a new accident-free container handling record of 20.43 20-ft equivalent units/hr (PORT, 1994).

The port is about 1.6 km (1 mi) from highway I-95 with travel through an industrial area. It is also served by a trunk railway.

Other Pertinent Port Information: The Port of Richmond has only one entrance which is controlled by a Pinkerton Guard on a 24-hour basis.

The Port of Richmond Terminal is in the midst of an $8-10 million expansion program involving a 96 m (315 ft) wharf extension, new gate entrance and maintenance building, plus upgraded container storage areas (PORT, 1994).

The likelihood of severe natural phenomena such as high winds and earthquakes are reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Richmond, the Uniform Building Code requires buildings to withstand wind speeds up to 120 km/hr (75 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 Census city population was approximately 203,000 and the population density was about 1,304 per km^2 (3,376 per mi^2). The distances to the five management sites on interstate routes are: Savannah River Site, 732 km (455 mi); Oak Ridge Reservation, 784 km (487 mi); Idaho National Engineering Laboratory, 3,900 km (2,426 mi); Hanford Site, 4,570 km (2,842 mi); and Nevada Test Site, 4,070 km (2,529 mi). Distances along rail routes are slightly longer.

Climatic Conditions

Richmond's climate might be classified as modified continental. Summers are warm and humid and winters generally mild. The mountains to the west act as a partial barrier to outbreaks of cold, continental air in winter, the coldest air being delayed long enough to be modified, then further warmed as it subsides in its approach to Richmond. The open waters of the Chesapeake Bay and Atlantic Ocean contribute to the humid summers and mild winters. The coldest weather normally occurs in later December and in January, when low temperatures usually average in the upper twenties and the high temperatures in the upper forties. Temperatures seldom lower to zero.

Precipitation is rather uniformly distributed throughout the year. However, dry periods lasting several weeks do occur, especially in autumn when long periods of pleasant, mild weather are most common. There is considerable variability in total monthly amounts from year to year so that no one month can be depended upon to be normal. Snow has been recorded during seven of the 12 months. Snowfalls of 10 cm (4 in) or more occur on an average of once a year. Snow usually remains on the ground only one or two days at a time. Ice storms (freezing rain or glaze) are not uncommon in winter, but they are seldom severe enough to do any considerable damage. The James River reaches tidewater at Richmond, where flooding has occurred in every month of the year, most frequently in March (28 times in the past 61 years), and only twice in July. Hurricanes and less severe storms of tropical origin have been responsible for most of the flooding during the summer and early fall. Damaging storms occur mainly from snow and freezing rain in winter, and from hurricanes, tornadoes, and severe thunderstorms in other seasons. Damage may be from wind, flooding, or rain, or from any combination of these (DOC, 1993c).

D.2.2.20 San Francisco, CA

San Francisco, CA, occupies the north portion of the peninsula forming the south entrance to San Francisco Bay. The Port of San Francisco, one of the largest ports on the bay, is the oldest and one of the most important on the Pacific Coast (DOC, 1992b).

San Francisco is a deepwater port stretching approximately 12 km (7.5 mi) along the southern and western shore of the San Francisco Bay. The approach to San Francisco and down the east side of San Francisco is open; however, there are restricted navigational areas. There is also considerable traffic in the bay area, and there is a traffic separation scheme under U.S. Coast Guard traffic control (Mitchell, 1994). Depths of 13.7 m (45 ft), or more, are available from the Golden Gate Bridge to most of the anchorages. Depths up to 12.2 m (40 ft) are available to most piers, including those at the container facilities in the vicinity of Islais Creek. The wide passage from the ocean to San Francisco Bay is reduced to approximately 1.13 km (0.7 mi) at the Golden Gate Bridge pier. The distance from the Golden Gate Bridge to the entrance of facilities near Islais Creek is approximately 19 km (12 mi)(DOC, 1992b). A map of the port is shown in Figure D-51.

The Port of San Francisco is under the control of the City and County of San Francisco, to which it was transferred by the State in 1969. The "authority" reports to an appointed board of Harbor Commissioners. The port is a multi-terminal, multi-function harbor complex that the Authority operates as a Landlord owner. Services of the port range from cargo handling along the southern waterfront to ferry terminals and tourism services — including a cruise ship terminal, ferry plaza, Embarcadero, excursion boat terminals, Fisherman's Wharf, and aquatic park located on the central, northern, and western sides of the port (POSF, 1993).

Principal container handling facilities are located at North Terminal (Pier 80), operated by Metropolitan Stevedore Co., and South Terminal (Piers 94-96) operated by Stevedoring Services of America. Breakbulk general cargo is handled at Piers 27 - 29 in the northern Embarcadero section of the city. Total tonnage handled in calendar year 1991 amounted to 5,994,000 metric tons (6,607,200 tons) and included 223,676 20-ft equivalent units of containerized cargo (AAPA, 1993). With the 1994 loss of four major container lines to Oakland and the closing of the Naval Supply Center, the Port's Chief Wharfinger expected container traffic for the year to drop to about 50,000 20-ft equivalent units (Mitchell, 1994).

South Container Terminal: This terminal has a total area of 30.6 ha (75.6 acres), one container freight station, and four gantry-type container cranes. Dock/quay lengths for cargo ships at South Terminal are three berths totalling 747 m (2,450 ft) in length. Depths alongside dock/quay are 12.19 m (40 ft) at mean low water. North Terminal has three 40.6 metric ton (45 ton) rail-mounted container cranes and two 30.5 metric ton (34 ton) rail-mounted container cranes (Jane's, 1992; AAPA, 1993).

North Container Terminal: This terminal has a square-shaped, finger-type pier comprised of 27.74 ha (68.6 acres), of which 13 ha (32.1 acres) are laid out for container operations. This terminal has five container cranes and its own CFS. Dock/Quay lengths for cargo ships at this terminal are four berths at a total length of 1,552 m (5,092 ft). Depths alongside dock/quay are 12.2 m (40 ft) at mean low water. Crane capacities at South Terminal include four 30.5 metric ton (34 ton) rail-mounted container cranes (Jane's, 1992; AAPA, 1993; Mitchell, 1994).

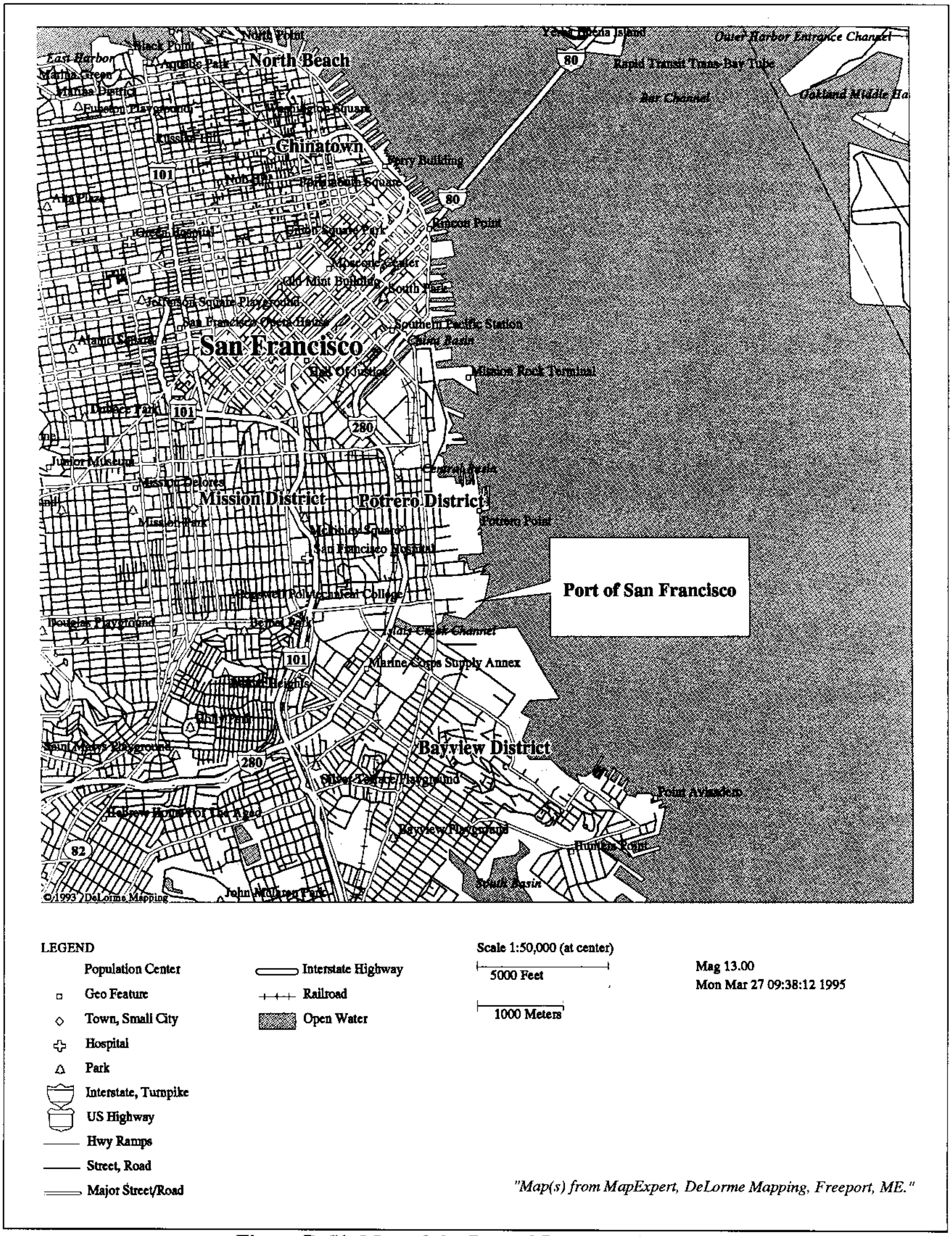


Figure D-51 Map of the Port of San Francisco, CA

Truck access to the container terminals is via Cargo Way and Third Street. Cargo Way connects South Terminal Piers 94 - 96 with 3rd Street. Entrance to the North Container Terminal (Pier 80) is at the intersection of 3rd Street and Army Street, which connects with I-280 and U.S. Highway 101 about 0.8 km (0.5 mi) from the entrance and about 1.6 km (1 mi) from the entrance to the South Container Terminal. These highways link up with the San Francisco/Oakland Bay Bridge (I-80) — the assumed route to Idaho National Engineering Laboratory and/or points east— which is roughly 1.6 to 2.4 km (1 to 1.5 mi) away. The Southern Pacific Railway serves both the North and South Container Terminals, and the Union Pacific also has tracks to the North Terminal's Pier 80. Trackage at South Terminal extends shipside parallel to the berth. Adjacent to the South Container Terminal is a 14.6 ha (36 acre) intermodal container transfer facility (Jane's, 1993; AAPA, 1993).

San Francisco has been served by a number of major container carriers. Lines calling at South Terminal include Grancolombiana and Evergreen. Liner companies using North Terminal include Blue Star Line, Central American Container Line, CSAV (Chilean Line), ELMA, Nedlloyd, NSCP, South Seas Shipping, and Splosna Plovba (Jane's, 1992; AAPA, 1993). However, in 1994, four of its five major container lines moved to Oakland (Adams, 1994; Mitchell, 1994).

Other Pertinent Information: Terminal security is the responsibility of the respective terminal operating companies. Facilities are fenced with controlled access and are patrolled by watchmen supplied by the International Longshoremen Workers Union. There are also City police officers permanently assigned for general port security (Mitchell, 1994). There are places within the container terminals for temporary segregation and storage of hazardous materials (Mitchell, 1993).

There are no regulations prohibiting the handling of containerized spent nuclear fuel. The port handles hazardous cargoes but, as far as known, has not handled spent nuclear fuel. The port allows Class A and B explosives in small amounts only (Mitchell, 1994).

All of San Francisco's marine terminals are located within the densely populated downtown area of the city and the large tourist population. Although there appears to be conflicting use of the Port of San Francisco's marine facilities (primarily attributable to its tourism business, much of which is centered to the north and west of the port's two container terminals) it is not deemed a major consideration. Terminal operators are responsible for accidents within their respective facilities. The Port Authority relies on the City of San Francisco's Fire Department hazardous materials team and the Coast Guard in case of an emergency. The City of San Francisco has a special Engine Company for responding to fires and other dangerous situations within port facilities, with about a five minute response time (Mitchell, 1993 and 1994). The Pacific Maritime Association handles hazardous materials instruction and training, and has just begun a program at the port (Mitchell, 1993). It is noted that U.S. Coast Guard statistics indicate that terminals in the San Francisco Bay have had only 31 reported collisions reported but an unusually high number of fires in recent years (21 fires reported between 1991 and 1993; the worst three-year fire record for major ports on the West Coast) (USCG, 1994b).

There are no known protected habitats or sanctuaries immediately near the terminals that might be affected by an accident in port. However, the predisposition of the City to severe earthquakes, and the high sensitivity of this area to protecting and maintaining environmental quality is considered a basis for concern. The city rests on the edge of the Pacific tectonic plate, while the opposite side of the Bay sits on the Continental plate. This results in the entire Bay area being a highly seismic zone. On April 18, 1906, San Francisco was the site of one of the largest recorded earthquakes in the contiguous United States, a Modified Mercalli Intensity XI (Bolt, 1978) due to movement along the fault line separating the two tectonic plates. The Uniform Building Code requires construction to withstand earthquakes and other severe natural phenomena (UBC, 1991). The Uniform Building Code requires construction for an acceleration of 0.40 g, the highest seismic ranking in the United States. High winds have not been a problem for the Bay, with a Uniform Building Code minimum basic wind speeds up to 140 km/hr [70 miles per hour (mph)].

The climatic and environmental conditions of the Port of San Francisco are the same as those reported for the Port of Oakland in Section D.2.2.15.

The 1990 population within 16 km (10 mi) of the port terminals was 1,265,529. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 1,060,000; Oak Ridge Reservation, 766,000; Idaho National Engineering Laboratory, 348,000; Hanford Site, 339,000; and Nevada Test Site, 461,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,570 km (2,803 mi); Oak Ridge Reservation, 4,130 m (2,567 mi); Idaho National Engineering Laboratory, 1,560 m (970 mi); Hanford Site, 1,420 km (882 mi); and Nevada Test Site, 1,160 km (722 mi). Distances along rail routes are slightly longer.

D.2.2.21 Seattle, WA

The Port of Seattle, WA, is located 230 km (143 mi) from the confluence of the Strait of Juan de Fuca and the Pacific Ocean. Seattle is located on Elliott Bay on the eastern shore of the Puget Sound, about 93 km (50 mi) south of the Strait of Juan de Fuca and about 5 km (3 mi) from the Sound. It is the largest and most important city in the Northwest, and one of the major ports on the Pacific Coast. Access from the Pacific Ocean is gained through the Strait of Juan de Fuca and Puget Sound. The transit from the Pacific Ocean to Seattle is open and considered relatively easy, with very deep waters during the entire approach to Seattle (DOC, 1992b). A map of the port is shown in Figure D-52.

The Port of Seattle is a large, diversified, multi-terminal port. Overall container tonnage for 1992 amounted to 7,510,000 metric tons (8,278,300 tons) and 1,155,000 20-ft equivalent units. It is managed by the Managing Director of the Marine Division and staff. Its facilities are municipally owned and leased to tenants (i.e., the Port Authority operates as a Landlord owner) (POS, 1994).

The port has five container terminals, of which two, Terminals 5 and 18, are considered public facilities:

Terminal 5: T5 is located on the West Waterway and is leased to and operated by American President Lines. Terminal 5 has a total area of 36 ha (89 acres), of which 24 ha (59 acres) can be used for container handling and storage. It has three container berths (Berths 4, 5, and 6), is equipped with six 50.8 metric ton (56 ton) Post-Panamax container cranes, and has two container freight stations. Terminal 5 has 760 m (2,500 ft) of marginal wharf, with 12.19 m (40 ft) of water alongside at mean low water. The terminal has good access to Interstate 5; about 3.8 km (2.4 mi) from the ramp to I-5 following a route entirely within the port's industrial district via North Marginal Way and West Seattle Freeway to South Spokane Street. I-5 is the principal north/south roadway linking Seattle with I-84 at Portland, OR (the assumed preferred, year-around route to Idaho National Engineering Laboratory) and/or I-90/82, which also links up with I-84 near Pendleton, OR. Terminal 5 is served by the Burlington Northern Railroad, whose tracks are located at the rear of the Terminal. The port is considering a proposal to provide Union Pacific service (Benham et al., 1994). Terminal 5 is served by major container lines including APL, OOCL, Star Shipping, and Westwood Shipping (Jane's, 1992; AAPA, 1993; D&B, 1993).

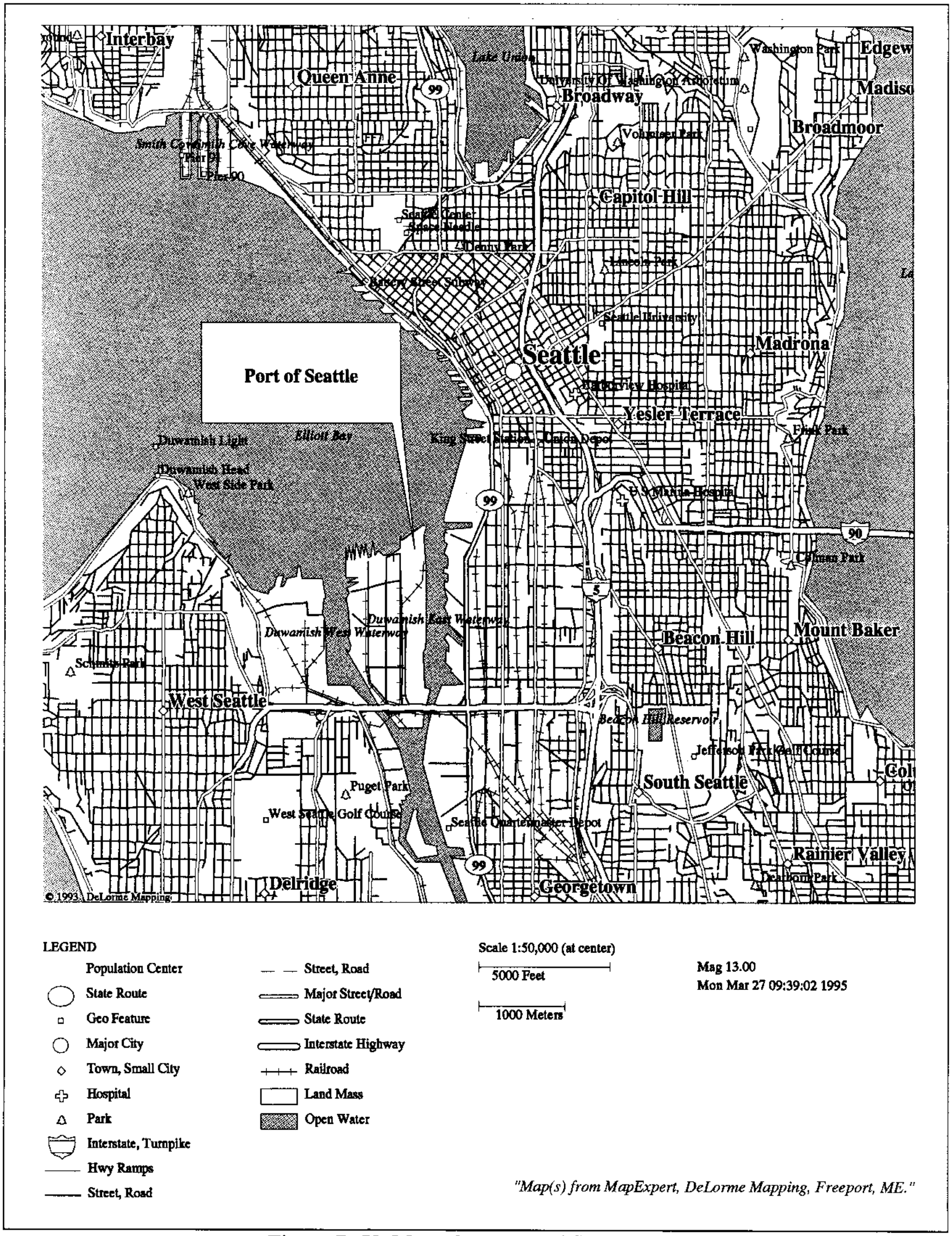


Figure D-52 Map of the Port of Seattle, WA

Terminal 18: T18 is located on the East Waterway (Berths 1 - 4, and 5 - 8), is operated by Stevedoring Services of America, and has a total area of 47 ha (116 acres) devoted to container handling and storage. It is also equipped with six, 40.6 metric ton (45 ton) container cranes and container freight stations. Terminal 18 has 1,844 m (6,049 ft) of marginal wharf, with 15.24 m (50 ft) of water alongside at mean low water. The terminal also has good access to Interstate 5; about 2.9 km (1.8 mi) from the ramp to I-5 following a route that is also entirely within the port's industrial district via South Spokane Street. Terminal 18 is also served by the Burlington Northern as well as the Union Pacific railroads via tracks along the wharf apron (i.e., ship-side).

Terminal 18 is served by several major container lines, including Barber Blue Sea, Grancolombiana Line, COSCO, d'Amico Line, Japan Line, Hyundai, Scindia Line, Chilean Line, ACL/CGM, and P&O Container Line (Jane's, 1992; AAPA, 1993; D&B, 1993; Benham et al., 1994).

Other Pertinent Information: There are potentially conflicting activities near the Terminal; petroleum products are pumped ashore at Terminal 5 (Berths 4 and 5), and across the East Waterway at Terminal 18 (Berths 2 and 3). The terminals are fenced with controlled access and guarded by watchmen on a 24-hour basis. There are areas within the container terminals for segregating hazardous materials cargoes.

The port's Emergency Response Plan relies on the City of Seattle Fire Department for hazardous materials response, with a technical support team including spent nuclear fuel handling experts from the DOE Hanford Site hazardous materials training for port workers is the responsibility of the individual terminal operators (Benham and Schuler, 1993; Benham et al., 1994). As noted in the accident information for Tacoma, the overall ship accident rates in the Puget Sound for the 1991-1993 reporting period are relatively low (USCG, 1994b).

Seattle's container terminals are somewhat separated from the City, which is generally north-east of the terminals. As already noted, these terminals have good access to Interstate highways without passing through congested city streets. However, T5 and T18 are both relatively close to some residential areas in West Seattle.

According to Mr. Schuler, Port Safety Officer, a port Commission resolution banning spent nuclear fuel shipments from the Port of Seattle has been in place for 3 or 4 years (Benham and Schuler, 1993). Reportedly, the Commissioners felt the Federal government was unresponsive to their requests for information concerning material being shipped and decided to ban further spent nuclear fuel shipment. As a result, the port no longer handles spent nuclear fuel and does not want it passing through its facilities. Since discussing this issue with the port safety official, DOE was informed that the Seattle City Council passed a resolution on December 8, 1993, which states the City's position that "high-level nuclear wastes should not be moved through Seattle or the Puget Sound by water or land transportation" (Noland, 1994). This issue is addressed in Section 6.5 of the EIS. Port officials (Benham and Schuler, 1993) thought the port had some prior experience with handling spent nuclear fuel, but this was not confirmed by available data going back to 1979 (NRC, 1993; SNL, 1994).

There are no known particularly environmentally sensitive areas (e.g., such as wildlife sanctuaries) in the immediate area of the terminals, but there is extreme public environmental sensitivity to potential environmental damage to the Puget Sound area.

The entire Puget Sound area is subject to severe earthquakes and volcanism. There have been two major earthquakes in the Puget Sound area this century; a Modified Mercalli Intensity VIII on April 13, 1949 and a Modified Mercalli Intensity VII-VIII on April 29, 1965 (Bott, 1978). On May 18, 1982, Mount Saint Helens suffered a major volcanic eruption (IPA, 1993). All the mountains along the Cascades Range, from Canada to Northern California, are volcanic in origin and are potentially active (Foster, 1971; Hamilton, 1976; IPA, 1993). The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Seattle, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a high seismic zone with an acceleration of 0.30 g.

The 1990 population within 16 km (10 mi) of the port terminals was 753,296. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 565,000; Oak Ridge Reservation, 395,000; Idaho National Engineering Laboratory, 122,000; Hanford Site, 62,900; and Nevada Test Site, 344,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Figures D-8 through D-17 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 4,670 km (2,900 mi); Oak Ridge Reservation, 4,240 km (2,636 mi); Idaho National Engineering Laboratory, 1,280 km (793 mi); Hanford Site, 360 km (226 mi); and Nevada Test Site, 2,130 km (1,322 mi). Distances along rail routes are slightly longer.

Environmental Conditions

A variety of aquatic species can be found in Puget Sound. Several animal species with special status may also be found in this area. A variety of marine mammals can be found in the central Puget Sound, including the Pacific harbor seal, California sea lion, killer whale, Dall porpoise, and harbor porpoise. In 1991, the U.S. National Marine Fisheries Services reported that the following endangered and/or threatened species may occur in the Puget Sound: the endangered gray whale, the endangered humpback whale, the threatened Stellar sea lion, and the endangered leatherback sea turtle (DOE, 1995). These species are not reported at the port. The U.S. Fish and Wildlife Service reported that the bald eagle and marbled murrelet, both listed protected species, may occur in the vicinity of the port (Frederick, 1994). Bald Eagles can be found throughout this coastal zone and American peregrine falcons are uncommon winter visitors (FWS, 1981a). The FWS's Ecological Inventory for the Puget Sound area indicates that the habitat of Elliott Bay is used by a variety of birds, including: shorebirds, gulls, sandpipers, turnstones, plovers, yellowlegs, herons, rails, great blue heron, waterfowl, loons, grebes, swans, geese, dabbling ducks, diving ducks, mergansers, American widgeon, pintail, mallard, seabirds, cormorants, alcids, common murre, and the pigeon guillemot. Adult concentrations of all of these species may be found in the Bay. Some of these species may also use this area as an overwintering area, a migratory area, and/or a nesting area (FWS, 1981a). It is also indicated that adult concentrations of Chinook salmon, coho salmon, and chum salmon are found in the West Waterway and Duwamish Waterway and use these water bodies and upstream segments as migratory and nursery areas.

According to the State of Washington's Department of Wildlife, the California sea lion uses the waters in the vicinity of Harbor Island as "haulouts" (i.e., areas regularly used by marine mammals for resting). Several seabird colonies also exist in this general area. There is a general lack of wetlands along the southeastern shore of Elliott Bay and along the East and West Waterways and the Duwamish Waterway (WDW, 1994a).

Climatic Conditions

The Strait of Juan de Fuca separates the northern coast of the State of Washington and the southern shore of Vancouver Island, Canada. Also in this general vicinity is the Port of Tacoma, Washington which is located 263 km (142 nautical mi) from the confluence of the ocean and the strait.

The city of Seattle is situated on a low ridge lying between Puget Sound on the west and the Green River valley on the east. The Olympic Mountains, which rise steeply from the Puget Sound are located approximately 80 km (50 mi) to the northwest. The mild climate of the Pacific Coast is modified by the Cascade Mountains and to a lesser extent by the Olympic Mountains. The climate is characterized by mild temperatures, a well-defined rainy season and prolonged cloud cover, especially during the winter months. The Cascades act as a very effective barrier in both winter and summer, shielding the region from both extreme cold and heat, respectively. The rainy season extends from October through March, with December accounting for the most rainfall. Approximately 75 percent of the annual total precipitation occurs during the winter rainy season. The dry season is centered around July and August. The majority of Seattle's precipitation is associated with normal, mid-latitude disturbances, which are most vigorous during the winter months. During summer, the dominant storm track (e.g., the polar jet) shifts northward into southern Canada, reducing the precipitation in the area. Summer thunderstorms do occur but do not contribute measurably to the annual rainfall budget. Prevailing winds are from the southwest, but occasional severe winter storms will produce strong northerly winds. Summer winds are generally rather light, with the occasional evidence of land-sea breeze effects creating northerly flows. Fog and low-level stratocumulus clouds form over the southern Puget Sound area in the late summer, fall, and early winter months, and often dominate the weather conditions of the early morning hours, reducing surface visibilities. Based on the 1951-1980 climatology, the first occurrence of freezing temperatures should occur around November 11, and the last incidence in spring around March 24 (NOAA, 1992g).

D.2.2.22 Wilmington, DE

The city of Wilmington, DE, sited on the Christina River, has large manufacturing interests. Both sides of the river at the city are lined with wharves that primarily support barge traffic. Deepwater facilities are located at the Port of Wilmington on the south side of the Christina River. The port is located about 3 km (2 mi) north of the Delaware Memorial Bridge on the left ascending bank of the Delaware River, approximately 100 km (62 mi) above the entrance to the Delaware Capes. The port is south of the city of Wilmington and is situated in an area of heavy industrial usage, which appears to be remote from residential, light business, and manufacturing areas (DOC, 1993c). A map of the city is shown in Figure D-53.

Access to the Port of Wilmington is gained via the Delaware Bay and Delaware River. The bay has natural depths of 15.4 m (50 ft) or more for a distance of 8 km (5 mi) from the entrance. A Federal project provides depths of 12.2 m (40 ft) past the entrance to the Christina River where the project depth is 10.6 m (35 ft). A traffic separation scheme has been established off the entrance of the Delaware Bay because of restrictions on passage through the bay and on up the Delaware River. Ships travelling to Wilmington must pass under the Delaware Memorial Bridge (DOC, 1993c).

The port is owned by the City of Wilmington. It is an "operating" port with stevedoring handled by two outside stevedoring companies. Principal cargoes are imported automobiles, dry bulk, roll-on/roll-off and refrigerated containers (primarily bananas and other tropical fruit) (POW, 1994). In 1993, the port handled about 936,000 metric tons (1,026,397 tons) of containerized cargo (about 100,000 20-ft equivalent units; AAPA, 1994). The port has 10,218 m^2 (110,000 ft^2) of chill/heat space and 36,806 m^3 (1,300,000 ft^3) of chill/freeze warehouse space. The terminal has two multi-purpose container cranes and one bulk cargo gantry crane. The marginal wharf area is 1,158 m (3,800 ft) long and there is a 155 m (510 ft) long floating roll-on/roll-off berth. Depth alongside the terminal at mean low water ranges from 11.58 m (38 ft) to 10.67 m (35 ft) due to silting. The port is equipped with one 40.6 metric ton (45 ton) multi-purpose container crane, one 29.1 metric ton (32 ton) multi-purpose container crane, and one 11 m^3 (14 yd^3) Clyde gantry crane (AAPA, 1993; Jane's, 1992; POW, 1994). Approximately half of the cargo going in and out

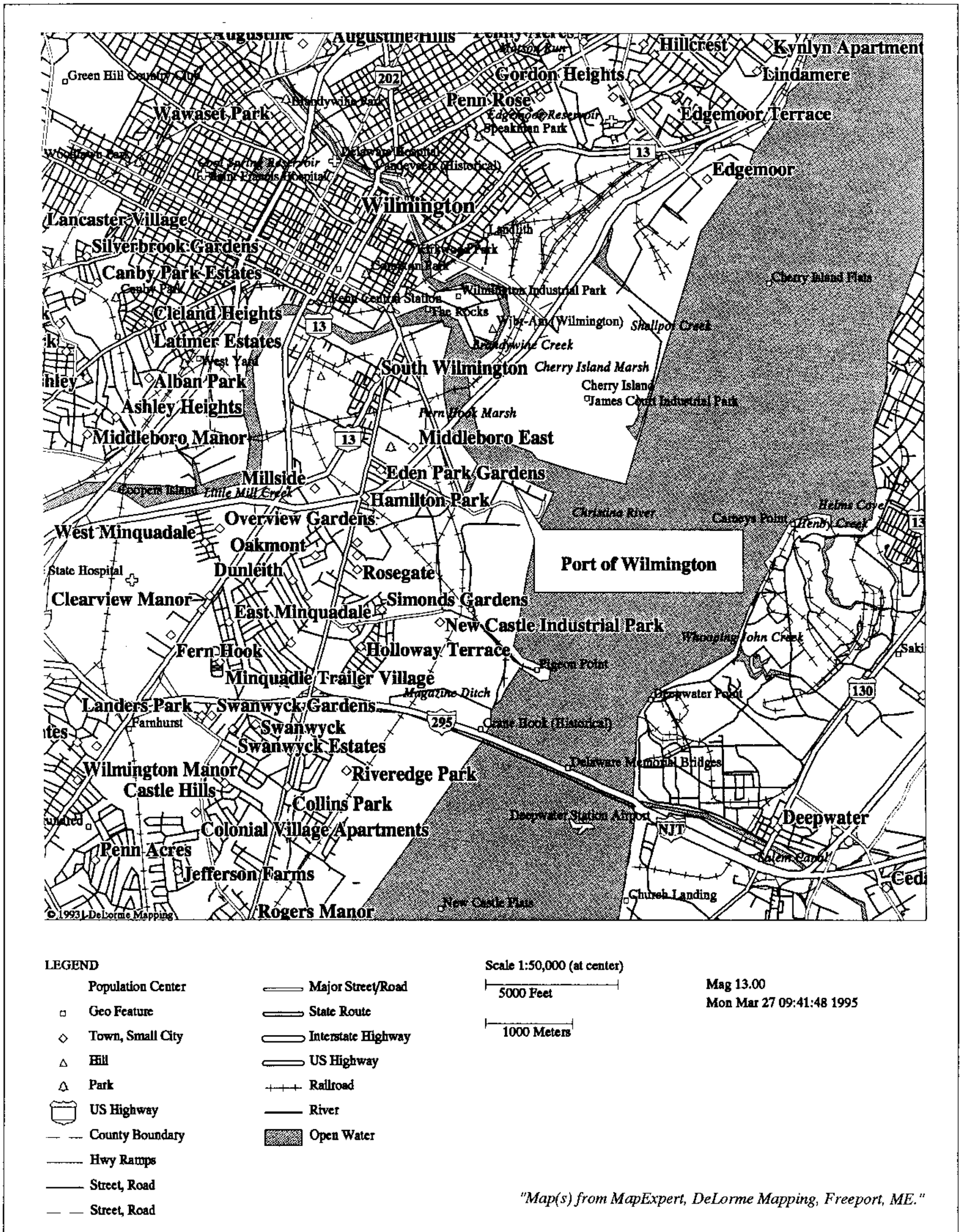


Figure D-53 Map of the Port of Wilmington, DE

of the port is food. Improved tropical fruit comprised 35 percent of the traffic in the port, while manufactured food products and finished perishables make up another 15 percent of the cargo traffic. Forest products handle 15 percent of the cargo traffic and imported steel makes up 7 percent. Several bulk commodities that are nonhazardous are the remaining 28 percent of the cargo handled by the port (Brooks, 1994).

The Port of Wilmington has direct access to I-495, a connector to I-95, which appears to be less than 1.6 km (1 mi) from the port and runs through the industrial district surrounding the Terminal. The Conrail and CSX railroads serve the port; it is not known if direct ship/rail transfer is possible.

Other Pertinent Information: Security of the general cargo terminals is maintained by the port police on a 24-hour basis. The wharves are fenced and truck access is via controlled terminal entrances. There is presently no place within the port for segregation and temporary storage of hazardous cargoes. There are no general cargo container lines currently serving the port, and there is no commercial container facility.

According to the Sandia National Laboratories' Radioactive Materials Postnotification Database, the port has not handled spent nuclear fuel since October 1984, when the database was initiated (SNL, 1994). There are no known conflicts with other hazardous materials in the immediate container terminal area. There are, however, chemical plants near the port, as well as a diversity of marine terminals and heavy tanker traffic (ship and barge) on the Delaware River. Other than increased risk of collision, these conflicts are not considered a major factor.

The port relies on the City of Wilmington's fire department for response in the event of a terminal hazardous materials accident. The port claims there is no hazardous materials training program and avoids handling hazardous materials (Casper, 1993).

There are no known protected habitats or sanctuaries near the port. However, at the mouth of the Christina River near the location of the port, there are extensive wetlands along the banks of the Delaware River.

The likelihood of severe natural phenomena, such as high winds and earthquakes, is reflected in the structural requirements for buildings in each area of the United States. These are shown in the Uniform Building Code (UBC, 1991). For the Port of Wilmington, the Uniform Building Code requires buildings to withstand wind speeds up to 130 km/hr (80 mph). The port is located in a low seismic zone with an acceleration of 0.075 g.

The 1990 population within 16 km (10 mi) of the port terminals was 381,502. The affected populations within 0.8 km (0.5 mi) of the interstate routes to the five potential DOE management sites are: Savannah River Site, 359,000; Oak Ridge Reservation, 297,000; Idaho National Engineering Laboratory, 535,000; Hanford Site, 584,000; and Nevada Test Site, 718,000. Populations along rail routes to these sites are slightly larger. These populations are shown in Tables D-7 through D-16 in Section D.1. The distances to the five potential sites on interstate routes are: Savannah River Site, 1,120 km (697 mi); Oak Ridge Reservation, 1,040 km (645 mi); Idaho National Engineering Laboratory, 3,890 km (2,416 mi); Hanford Site, 4,560 km (2,832 mi); and Nevada Test Site, 4,160 km (2,588 mi). Distances along rail routes are about the same.

Environmental Conditions

The Port of Wilmington is located within Zone 5 (Delaware Estuary/Bay) of the Delaware River. Protected water uses for Zone 5, which encompasses River Mi (RM) 48-79, are water supply (industry), wildlife, resident fish propagation and maintenance, anadromous fish passage, primary contact, and navigation (DRBC, 1994). However, within Zone 5, fish and other aquatic life are currently impacted due to low dissolved oxygen levels from point and nonpoint source discharges. Further south in Delaware Bay (Zone 6), shellfish consumption is an impaired use due to bacterial infestations from local point and nonpoint sources.

The Delaware River at Wilmington is classified as a low salinity estuarine (generally 0.5 to 5 ppt) and tidal freshwater habitat. Aquatic organisms typically found in the waters of this area include: American shad, atlantic sturgeon, American eel, blueback herring, shad, alewife, white catfish, brown bullhead, perch, striped bass, bluegill, crappie, pumpkinseed, largemouth bass, carp, and chain pickerel (FWS, 1980f). In addition, the Delaware River is used as a migratory area by the shortnose sturgeon, a Federally listed endangered species. South of Wilmington, the shoreline of the Delaware River becomes much less developed and numerous fish and wildlife management areas and wetlands are found along the lower Delaware River and Bay. Bald eagles are found throughout these areas. The Delaware Bay supports high densities of geese and ducks along the shores. Waterfowl, particularly loons and grebes, and seabirds, particularly gannet, Wilson's petrel, and greater shearwater are also found in the Bay area. Osprey, peregrine falcon, and Cooper's hawk migrate in fall along the Delaware Bay to Cape May Point. High densities of whitetail deer also occur along the shore of the Bay (FWS, 1980f).

The Delaware Natural Heritage Inventory reported that "there are no Species of Special Concern within 0.8 km (0.5 mi) of the Port of Wilmington" (Dalton, 1994). However, the Inventory reported that the Peregrine falcon (Federally listed endangered) has nested 2.4 km (1.5 mi) from the port, and the short-nosed sturgeon (Federally listed endangered) is found in the Delaware River. It was the opinion of the Inventory that "these species would not be affected by normal operations at the port."

Climatic Conditions

Geographically, Delaware is part of the Atlantic Coastal Plain, which consists mainly of flat lowland and marshes. Small streams and tidal estuaries comprise the major drainage systems for the State. The Delaware River, Delaware Bay, and Atlantic Ocean form the eastern border of the State, while the Chesapeake Bay is the western boundary [approximately 56 km (35 mi) to the west]. These large water bodies contribute significantly to the climate of the Wilmington, DE, region.

Generally, summers are warm and humid and winters are considered rather mild. Summer temperatures rarely exceed 100°F, and average daily temperature during January (the coldest month) is 32°F. The majority of winter precipitation falls as rain, but precipitation during the winter months is often mixed rain, snow, and sleet. However, frozen precipitation rarely remains on the ground more than a few days. The proximity of the water masses causes humidity to remain relatively high year-round, which causes frequent fog events. Light southeasterly winds (e.g., off the Delaware Bay) tend to be most favorable for fog formation, while north-northeast winds tend to transport industrial pollution from the Philadelphia metropolitan area into the region. Rainfall distribution is fairly uniform throughout the year, but the greatest amounts normally come during the summer months in the form of thunderstorms. During the fall, winter, and spring seasons, the majority of rainfall is associated with extratropical and tropical cyclones track along the eastern seaboard of the United States. Hurricane-force winds are rarely experienced in the Wilmington, DE, region. However, strong south and southeasterly winds can cause high tides in Delaware Bay and the Delaware River, causing lowland flooding and damage to bayfront and riverfront properties (NOAA, 1992k).

D.3 Main Routes

The routes selected for potential marine transport are discussed in Appendix C. These routes cover the transport of the spent nuclear fuel from the country of origin to the first port of call in the United States. In the port incident-free and accident analysis it has been assumed that the vessel carrying the spent nuclear fuel would not unload the material at its first port of call. Intermediate port calls have been assumed in the analysis. In the marine impact accident and incident-free analysis, the intermediate port calls result in additional travel time which has been incorporated into both analyses. In the port analysis, this results in additional workers who could be affected by incident-free impacts and additional locations where accidents could occur. Due to the large variability associated with the movement of the vessel between U.S. ports, no specific route has been identified for use in the analysis. With the approach used in this analysis, the specific routes used between the U.S. ports would not affect the results of the risk assessment.

D.4 Accident-Free Impacts: Methods and Results

D.4.1 Introduction

This section of the appendix provides an overview of the approach used to assess the risks associated with port activities involved in transferring the spent nuclear fuel from the vessel to a vehicle for transport to the management site. Included here is a discussion of the incident-free risk assessment methodology and the results of the analyses, including an assessment of the cumulative risk associated with the marine transportation of the foreign research reactor spent nuclear fuel through U.S. ports.

The risk assessment results are presented in terms of a per shipment risk, annual risks from incident-free transport, as well as for the total risks associated with the program.

D.4.2 Scope

All foreign research reactor spent nuclear fuel shipments that would require ocean transport are expected to occur via one of four types of vessels: container ships, roll-on/roll-off vessels, general cargo (breakbulk) vessels, or purpose built vessels. In the incident-free analysis, it has been assumed that all shipments are made on either a breakbulk or a container vessel, an assumption intended to provide bounding assessments of the risks associated with port activities required for the transfer of spent nuclear fuel.

D.4.2.1 Nonradiological Risk of Marine Transportation Related Activities

This portion of the risk assessment is limited to estimating the human health risks incurred during spent nuclear fuel unloading and handling during port operations at U.S. ports and during the vessel's approach to the port and movement within the port. The nonradiological risks from these activities were assessed as resulting in a negligible impact on the health of the public and workers. Approximately 56,000 port calls involving vessels engaged in foreign trade are made at U.S. ports every year (DOC, 1994). As discussed in Appendix C, each of these vessels has the capacity to carry hundreds of pieces of cargo of the size of a container carrying a spent nuclear fuel transportation cask (typically, container vessels carry between 800-1,000 containers, while some carry many more). This translates to millions of pieces of cargo every year. To fulfill the needs of the basic implementation of Management Alternative 1 of the proposed action, less than 60 transportation casks would need to be shipped per year. This is less than 0.001 percent of the total number of pieces of cargo (originating in foreign countries) to be handled at U.S. ports each year. The limited number of shipments per year should not result in a significant change to the risks to the public including the port workers.

D.4.2.2 Radiological Risks of Marine Transportation

The risks that result from the radioactive nature of the shipments are addressed for both incident-free transportation and accident conditions. The radiological risks associated with the incident-free shipping conditions result from the potential exposure of members of the crew and dock workers to external radiation in the vicinity of the packaged fuel. No other exposure is considered, due to the relative isolation of the material from the general public during all phases of the port activities associated with the transfer of the spent nuclear fuel from the ocean going vessel to the overland transportation mode.

All radiologically-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent (EDE) from the external radiation exposure and the 50-year committed effective dose equivalent from internal radiation exposure. The EDE is the sum of the tissue and organ weighted dose equivalents for all irradiated tissues and organs. The committed effective dose equivalent considers the initial exposure and the effects of radioactive decay and elimination of the radionuclide through ordinary metabolic processes over the 50-year period. Radiation doses are presented in units of person-rem for collective population and rem for individuals. The impacts are further expressed as health risks, primarily in terms of latent cancer fatalities (LCF). The health risk conversion factors were derived from International Commission of Radiological Protection Publication 60 (ICRP, 1991).

D.4.3 Port Facility Operations

This section describes the principal activities that are performed at a port facility to transfer a radioactive material package ("cask") from an ocean vessel to a surface conveyance, such as a truck trailer or railcar. The purpose of this description is to assist in establishing an estimate of the ionizing radiation dose to personnel that could be associated with the port intermodal transfer. The description of activities, and estimates of durations of specific tasks and personnel requirements is presented later in this section.

The off-load operation would take place at a "facility of particular hazard," as defined in 33 CFR 126.05, that is designated by the Captain of the Port. The Captain of the Port is a U.S. Coast Guard officer that enforces, within his/her respective port, safety, security and marine environmental protection regulations. These include, without limitation, regulations for the protection and security of vessels, harbors, and waterfront facilities; anchorages; security of vessels; waterfront facilities; security zones; safety zones; regulated navigation areas; deepwater ports; water pollution; and ports and waterway safety. The Captain of the Port designates and permits "facilities of particular hazard."

Such a facility is allowed to handle "cargoes of particular hazard" including "highway route controlled quantities of radioactive material," which includes spent nuclear fuel. The Captain of the Port could establish a safety zone or security zone around the vessel, if necessary. These zones would prohibit unauthorized personnel from entering the area. Usually a "facility of particular hazard" will have a secured area onsite for the storage of "cargoes of particular hazard." This facility would be used for the temporary storage of spent nuclear fuel, if necessary. Usually, these cargoes are loaded on a truck or train that departs for its destination soon after being checked by a facility employee and inspected by the proper authorities.

Each "facility of particular hazard" has an operations manual that outlines procedures for handling "cargoes of particular hazard," the personnel used and their qualifications, emergency procedures, and contact numbers. Only the Captain of the Port can approve the required operations manual, and only the Captain of the Port can approve any changes made to the operations manual. The content of the operations manuals can vary by port location and size, and by the type of materials handled. The operations manual of the facility under consideration for off-load operations should be studied prior to receipt of any spent nuclear fuel.

D.4.3.1 Intermodal Transfers

The intermodal transfer of the container (or cask) is largely a mechanical lifting operation with somewhat limited personnel participation. Unloading of vessels is generally performed by members of the International Brotherhood of Longshoremen (East Coast and Gulf Ports), or the International Longshoremen and Warehouseman Union ("Longshoreman") (West Coast ports), sometimes with support from the vessel's crew.

There are various configurations of container (or cask) storage aboard ship that could arise. However, as a preference, containers (or casks) are transported below decks. The following sections describe the principal operations that must occur to achieve both transfer of the container (or cask) from the ship, and to prepare it for departure from the port. It should be noted that as a general rule, departure from the port occurs as soon as is practicable, since the intermodal transfer is merely part of an "in progress" transportation activity, and radioactive materials transport should be expeditious. Infrequently, containers (or casks) may be (temporarily) stored at port facilities for some reason, such as bad weather.

D.4.3.1.1 Container Transfer to Truck Trailer or Railcar

If the port routinely receives containerized freight, it will be equipped with a crane adapted to handle containers. These cranes use a spreader bar equipped with International Standards Organization twistlocks at each of its four corners. The length of the spreader bar is automatically adjustable to accommodate the two International Standards Organization standard container lengths of 6.1 m (20 ft) or 12.2 m (40 ft). Casks are normally shipped in the 6.1-m (20-ft) containers. The twistlocks mate with standard fittings in the corner posts of the container, and are automatically actuated by the crane operator to attach the spreader bar to the container. Typically, no personnel are on the container when the spreader bar is attached. Engagement can be verified by the crane operator or, depending on the container stacking arrangement or port practice, by Longshoremen on the deck. The crane operator is in an enclosed cabin and is usually separated by a considerable distance from the cargo. The procedures described below apply to so-called cellular container ships or combination container/breakbulk ships.

Once engaged, the container is lifted from the hold of the ship, up and over the side to a container trailer, or railcar, on the dock. Engaging the container and moving it to the transporter, takes about 1.5 minutes on average (about 45 containers per hour).

The routine unloading is to install the container on a standard over-the-road container trailer which is pulled by a specially made tractor used at ports. These dock tractors have a single person cab and a hydraulically driven "fifth-wheel" which is used to raise the front end of the container trailer much higher than it would be for regular transport. This allows the Longshoremen to move the container trailer without having to raise and lower the trailer front landing gear at each re-positioning of the trailer. The dock tractor then moves the container to a freight staging area, parks it, connects to an empty container trailer, and re-positions under the container crane. Usually, several dock tractors are used to continuously move containers from "under the hook." Dock tractors are not suitable for over-the-road use.

The receiver (or the agent for the receiver) generally arranges with the Longshoremen to install the cask container directly onto the container trailer, or railcar, which will be used for overland transport, and which has already been inspected. The container trailer will be pulled by the tractor which is to be used for transport.

If the containerized cask is placed on a dock container trailer, sometimes called a "bombcart," then it must be later moved to the trailer which is to be used for transport. This transfer can be made using a large, industrial fork lift, top lift, or a small mobile crane ("forklift") specifically designed to move containers in the port freight staging areas. A bombcart is a special container trailer, used only within the port facility, that does not have twistlocks at its four corners to secure the container being loaded or unloaded.

Spotting the container on its designated trailer (or railcar) and securing it using the trailer mounted International Standards Organization locks, requires two (2), or four (4) longshoremen (at each end of the trailer) and takes about 30 seconds. Four (4) longshoremen have been used for this task at some ports. Once the container has been loaded onto its trailer, it moves immediately away from the container unloading area to a staging area so that ship unloading can continue. The staging area is established by port authorities, but must be approved by the Captain of the Port.

The staging area is usually close to the container unloading area, on the port property, and may be an area where hazardous materials are routinely handled. It may be an indoor location, such as a warehouse. It is used for the conduct of any inspections or surveys that may be desired, to verify documentation received from the ship's captain, to verify marks and labels on the containers, to verify securement of the load, to assemble required documentation for the overland portion of the transport, and install or verify placards. (It should be noted that foreign origin shipments are prepared in accordance with International Atomic Energy Agency standards, which are generally compatible with NRC and the Department of Transportation regulations. In accordance with International Atomic Energy Agency regulations, containers usually are prepared with an oversized label, which is an International Atomic Energy Agency permitted substitute for placards. Even if placarded, the placards usually do not conform to the "Highway Route Controlled Quantity" placard used for these types of shipments in the United States. The overland portion of the transport leaves from this area. Inspections are described in Section D.4.4.

The National Defense Authorization Act for Fiscal Year 1994 requires that, to the extent practicable, casks containing spent nuclear fuel should be moved expeditiously from the port. However, infrequently, continuation of the transport may not occur immediately. This may be due to unplanned events such as severe weather, equipment breakdown or inspection discrepancy, or to planned actions such as queuing of the receipt of individual containers at the receiving site. If one or more containers must remain at the port, they are normally moved to a bonded warehouse, with the container remaining on its transporter. The warehouse is considered a secure area, and it typically meets the requirements of a "safe haven."

Specific handling for rail shipments depends upon the location of rail track with respect to the container handling crane "foot print." If the rail line is within the foot print, then containers are loaded directly onto the railcars and secured using International Standards Organization locks in the deck of the railcar. Typically, two containers are loaded onto each railcar. If the rail line is not in the foot print, then the container is loaded onto a dock container trailer and moved to the rail line. An industrial forklift is used to transfer the container to the railcar. Railcars may be moved by a switch engine, but more commonly, a railcar tugger is used.

For spent fuel shipments, the railcars carrying loaded containers are separated from each other by buffer cars. These cars are usually empty gondolas or flat cars. A caboose is usually provided for escorts and required security equipment. The buffer cars are selected so that the escorts can have a good view of the container cars. Containers mounted on container trailers are not shipped on the railcar in a "piggyback" configuration because of concerns related to the resulting high center of gravity.

D.4.3.1.2 Container Transfer Using Jib-Type Cranes

The port may not have a container crane and instead rely on a dockside, pedestal mounted, or ship installed, jib crane. Containers are moved using this type of crane by attaching a four-legged sling to the crane hook, and extending one leg of the sling to each of the four corners of the container. The sling must be manually attached to (and later removed from) the International Standards Organization fitting at the top of the corner posts of the container. The attachment and removal is done by two longshoremen, who must climb on top of the container.

The attachment of the sling can take as long as three minutes. The reason for this is that, typically, the longshoremen climb onto the container before the crane operator has positioned the crane and lower the sling for attachment. The longshoremen also provide hand signals to direct the positioning for the crane. Disconnecting the sling from the container is done more quickly, and it is usually not necessary to climb onto the top of the container. Two longshoremen usually lock the container to the container trailer and disconnect the sling, but sometimes four are used.

If the ship is equipped with a jib crane, it may also be used to remove containers. The process is the same as with a dock mounted crane, but the crane is operated by a member of the ships crew. Except for the operation of a ship mounted crane, members of the ships crew do not generally have a role in the unloading of the ship.

D.4.3.1.3 Roll-on/Roll-off Operations

In the roll-on/roll-off configuration the casks (either containerized, freestanding, or palletized) are already on the trailer that is used for overland transport. After unlashing, the trailer is moved to the staging area by a longshoreman using a dock tractor.

Unlashing of the trailer may involve up to four longshoremen, and require up to 5 minutes. Transfer of the trailer to the staging area can take as long as 15 minutes depending on the ship's hold and ramp conditions and the distance to the staging area. After the trailer is spotted in the staging area it is connected to the tractor that is used for over-the-road transport.

Since the trailer has not been available for inspection, if an inspection is required [other than that done by the tractor driver(s)], it is performed at the staging area. If the trailer is foreign owned, temporary apportioned motor vehicle tags are provided by the receiver or receivers' agent.

D.4.3.1.4 General Cargo Operations

Breakbulk operations could involve either a containerized or free standing cask. Typically, a free standing cask is mounted on a pallet to facilitate the handling of the cask using the cranes and winches commonly found on ships and at dock side. Handling of a containerized cask would follow the same operation described in Section D.4.3.1.1.

Breakbulk cargo handling of a free standing cask is more labor intensive, since the cask must be unlashed from the deck and may have to be moved using winches to a hatch opening. A crane is used to lift the cask out of the hold and onto the dock. Up to 4 longshoremen may be used to move the cask in the hold and attach crane rigging to the cask or pallet. Two (2) or more longshoremen may be required to complete the transfer to the dock. At the dock, the pallet is typically placed on a standard flat bed trailer and secured with chains or other binders. Total handling time is less if the cask is transported in the center of the hold, as it likely would be if a chartered vessel were used.

In general, breakbulk cargo requires the longest unloading times, compared to containerized freight and roll-on/roll-off configurations. While a good unloading time for general cargo is about 5 minutes per crane load, radioactive materials transfer can take as long as 20 minutes if the cask is not transported on a pallet and must be rigged separately.

Breakbulk shipment of free standing spent nuclear fuel casks is perceived to result in a somewhat less reliable tiedown of the cask to the deck of the vessel. There is also an increased risk of damage to the cask or its pallet due to the variability in lift fixtures on each pallet. For these reasons breakbulk shipments of spent fuel casks have not been routinely made since the mid 1970's. This mode of shipment is not expected to be routinely used for the transport of spent nuclear fuel, except as it would apply to the use of purpose-built ships.

D.4.3.2 Key Intermodal Tasks and Task Durations

This section summarizes the key intermodal handling tasks, and estimates the personnel requirements and task durations for the transfer of the casks from the vessel to the land conveyance. These summaries are based on the narratives presented previously. Actual handling times and resource requirements can be widely variable, depending in large degree upon the cask configuration, transport vessel, intermodal handling equipment, port practice, and specific procedures which could be implemented for a given shipment or shipping program.

Port inspections are described separately in Section D.4.4.

D.4.3.2.1 Intermodal Handling of Containerized Casks

Ports equipped for intermodal handling of containers have achieved average rates of transfer of general cargo containers between the vessel and dock of 45 per hour, or about one container each 80 seconds. This rate may not be achieved for containers carrying spent nuclear fuel. For conservatism, a transfer time of 2 minutes per container is assumed. Longer transfer times would be expected if the port is not equipped with container cranes. A transfer time of 3 minutes is assumed if jib or boom type cranes are used with slings to lift the containers. Containers are assumed to be installed on the container trailer which would be used in over-the-road transport.

Port practices, such as union rules and safety procedures, would dictate the number of personnel used to unlash, transfer, and lash the container to its transporter. Consequently, the number of personnel required for each task could vary slightly between ports.

Each shipment, consisting of one or more containers, is expected to be observed by one or more persons who represent various interests in the shipment. These observers would have no active role in the transfer of the container, and would be expected to be 9.1 m (30 ft) or more away from the container.

Vessel crew members do not normally participate in container transfer operations, except for a member having responsibility for the cargo. Only this individual is considered to be present during transfer, stationed at the vessel hatch.

Table D-8 summarizes the handling of a container on a container ship. All of the distances are assumed to be from the container surface, or the projected container surface if an open container is used. There are no tasks which require contact with the cask surface.

Table D-8 Container Transfer Summary

Task / Personnel/Location	Unlash Cargo	Attach to Crane[a]	Transfer to Dock	Lash to Transporter	Move to Staging
0-.9 m (0-3 ft)	2 - 4[a,b]	1[b]	-	2 - 4[b]	-
duration (min)	0.25	0.5	-	0.25	-
1.5-3 m(5-10 ft)	-	-		-	-
duration (min)		-	-	-	-
3-6 m (10-20 ft)	-	-	-	1[c]	1[c]
duration (min)	-	-	-	0.25	3
6-9 m (20-30 ft)	1[b]	2[d]	1[d]	1[d]	-
duration (min)	0.25	0.5	1	0.1	-
9 m (30 ft)	1[e]	1[e]	1[e]	4[f]	4[f]
duration (min)	0.25	0.5	0.1	0.25	0.25

[a]*Crane attachment to containers is automated.*

[b]*longshoremen*

[c]*truck driver*

[d]*crane operator*

[e]*ships crew*

[f]*observer*

Containerized casks could be shipped aboard container or general cargo vessels. No significant difference in transfer times is expected between these vessel types.

D.4.3.2.2 Intermodal Handling of Roll-on, Roll-off Casks

Casks in a roll-on/roll-off configuration, either containerized or palletized are assumed to be transported on a roll-on/roll-off vessel and received at a port equipped to support roll-on/roll-off operations. Assumptions regarding port practices, observers and crew members are the same as those made for containerized or palletized cask transfer.

Removal of the trailered cask from the vessel is assumed to be done using a port tractor. Attachment of the trailer to the tractor which would be used for over-the-road transport must be done in the freight ready area, or the staging area.

All of the distances are assumed to be from the trailer or personnel barrier surface, or the projected trailer surface if there is no personnel barrier. There are no tasks which require contact with the cask surface.

Table D-9 summarizes the cask unloading and transfer activities for a roll-on/roll-off cargo vessel.

Transfer of roll-on/roll-off configured casks is not expected to occur on vessels not equipped with a ramp. Consequently, lifting of the trailered cask by crane is not expected to occur.

Table D-9 Roll-on/Roll-off Cask Transfer Summary

Task Personnel/Location	Unlash Cargo	Attach to Crane	Transfer to Dock	Lash to Transporter	Move to Staging
0-.9 m (0-3 ft)	4[a]	2[a]	-	4[a]	-
duration (min)	4	0.5	-	0.5	-
1.5-3 m (5-10 ft)	-	-	1[b]	-	-
duration (min)	-	-	0.25	-	-
3-6 m (10-20 ft)	1[a]	1[b]	1[b]	2[b]	1[b]
duration (min)	4	0.5	2	0.5	3
6-9 m (20-30 ft)	1[c]	1[c]	1[c]	-	-
duration (min)	4	0.5	0.25	-	-
9 m (30 ft)	-	-	-	4[d]	4[d]
duration (min)	-	-	-	0.5	0.25

[a] *longshoremen*

[b] *truck driver*

[c] *ships crew*

[d] *observer*

D.4.3.2.3 Intermodal Handling of Free-Standing (Palletized) Casks

As previously noted, casks are expected to be mounted on a skid, cradle or pallet ("pallet") to facilitate handling, lifting, and stowage. Transfer of these casks is usually somewhat more labor intensive than handling containerized casks, since the pallets are not standardized. The pallets are usually uniquely designed to accommodate a specific cask. Consequently, more effort is usually required to secure the cask in stowage, and to install lift slings for transfer. In addition, some care is needed to ensure that lifting and handling operations do not damage the cask.

Assumptions regarding port practices, observers, and crew members are the same as those made for containerized cask transfer.

It is assumed that the palletized cask would be installed on a flat bed trailer not necessarily having the tiedown fixtures required to secure the pallet. Some additional effort is expected to be required to secure the pallet to a trailer, compared to that required for containerized casks. However, it is assumed that the pallet is placed on the trailer that would be used for over-the-road transport so that no subsequent transfer of the pallet is needed.

Table D-10 summarizes the palletized cask unloading and transfer activities for a breakbulk cargo vessel. Distances are from the edge of the pallet, or its projected edge. There are no tasks which require contact with the cask surface.

D.4.4 Port Inspection Activities

There are several agencies, both Federal and State that could make an inspection of the cargo at any point from when the vessel docked while the cargo is still on board, until the cargo reaches its final resting place in the facility. The U.S. Coast Guard has recently designated personnel to inspect hazardous cargoes, specifically containers laden with hazardous cargo. The U.S. Coast Guard, however, has no current programs in place for the training of inspectors of radioactive materials. This may change in the near future. The U.S. Coast Guard does have an aggressive program for container inspection and compliance. The U.S. Coast Guard would perform an inspection on the vessel, including all documentation (bills of

Table D-10 Palletized Cask Transfer Summary

Task / *Personnel/Location*	*Unlash Cargo*	*Move to Hatch*[a]	*Attach to Crane*	*Transfer to Dock*	*Lash*[b] *to Transporter*	*Move to Staging*
0-.9 m (0-3 ft)	4[c]	0 - 4[c]	2[c]	-	4[c]	-
duration (min)	4	0 - 5	0.5	-	4	-
1.5-3 m (5-10 ft)	-	-	-	-	-	-
duration (min)	-	-	-	-	-	-
3-6 m (10-20 ft)	-	-	-	-	-	1[d]
duration (min)	-	-	-	-	-	3
6-9 m (20-30 ft)	2[c,e]	0 - 2[c,e]	2[c,e]	1[c,e]	4[c,e]	4[f]
duration (min)	4	0 - 5	0.5	0.1	4	0.25
9 m (30 ft)	-	-	1[g]	1[g]	1[g]	-
duration (min)	-	-	0.5	2	0.5	-

[a]*this task is not required if the cask is in the center of the ships hold*

[b]*transporter is to be used for over-the-road transport*

[c]*longshoremen*

[d]*truck driver*

[e]*ships crew*

[f]*observer*

[g]*crane operator*

lading and dangerous cargo manifests) and container placarding. Once the cargo is off-loaded, NRC may require an inspection of the container or cask and perform a radiation survey. Also, state agencies that are designated with such responsibilities as safety and transportation may require an inspection, especially on the tractor and semi-trailer transporting the casks. These latter inspections could take place dockside, at the facility, at a staging area, or at the gate area of the port. It is also possible that there would not be any inspections made by any agency.

The principal kinds of inspections that normally occur are: (1) verification of container (or cask) marks and labels to the accompanying documentation; (2) verification of radiation readings around the container (or cask); and (3) inspection of the transport vehicle, typically a tractor-trailer rig. Other inspections, such as condition of a container, can also be performed. Most of the inspections performed are done at the staging area, although inspection on the ship is also possible.

Port inspections are discretionary in that there is no regulatory requirement that they be performed by any party, with two exceptions. One exception is that a radiation survey map must be prepared for overland transport by truck and rail. This map must show the radiation levels at 2 m (6.6 ft) from the container or cask, and it must show the radiation level in the truck normally occupied by the driver. The agent for the receiver normally completes this map. A second exception is that State laws may require a permit for the transport of the spent fuel. Typically, this permit requires an inspection of the transporter for road worthiness, and sometimes a review of other documents. Inspections of railcars are normally not done by state inspectors. The performance of additional inspections may be established by (local) policy, procedures, or preference. In this context, inspections may occur more than once. The reason for this is that Federal agencies, such as the Department of Transportation and the U.S. Coast Guard, and the States (and the port authority), have a right of inspection. For any given shipment or individual cask, those agencies may not be represented, and even if represented, the right of inspection may not be exercised.

The representative of the receiver normally verifies that the marks and labels of the container conform to the documentation supplied by the shipper, that radiation levels are within U.S. regulatory limits, and that they conform to the radiation survey documents supplied with the shipping papers. These verifications are usually made after the container is removed from the ship and is in place on its transporter. Surveys of the container can also be performed aboard ship. This may be done for example, if there was a belief that actual radiation readings could be higher than those reported in the shipment documentation because of some event that occurred in transit, or for information.

Inspections of the transport equipment may be required by the State. These inspections are normally done prior to loading of the container on the bed of the trailer or railcar. This ensures that the container is loaded on an acceptable transporter. There is no radiation exposure which is attributable to this inspection. Verification of container tiedown is performed by the truck driver, or rail crew, as required by current regulations. Typically, tiedowns are also verified by a representative of the consignee. Tasks and personnel requirements are summarized in Table D-11.

Table D-11 Summary of Inspection Tasks and Personnel Requirements Per Container[a]

		Federal Agencies[b]					
		USCG	*DOT*	*NRC*	*State*	*Local/Port*	*Receiver*
Container	Personnel	1	1	1	1	1	1
	Time (min)	5	2	2	2	5	5
Roll-on/Roll-off	Personnel	1	1[c]	1	1[c]	1	1
	Time (min)	2	15	10	15	5	5
Breakbulk	Personnel	1	1[c]	1	1[c]	1	1
	Time (min)	2	15	10	15	5	5

[a]*Personnel expected to be within 3 m (10 ft) of the container.*

[b]*Discretionary inspections which may be performed; USCG = U.S. Coast Guard, DOT = Department of Transportation.*

[c]*Includes trailer inspection.*

D.4.5 Port Worker Incident-Free Analysis Methodology

Incident-free impacts of the offloading of foreign research reactor spent nuclear fuel have been estimated for port workers, inspectors, and observers of the activity. It has been assumed that no member of the public, other than the above-mentioned workers, would be present at the port during offloading. Ports tend to be relatively large areas with little or no access by the general public. Impacts of the incident-free shipment of foreign research reactor spent nuclear fuel on the general public would not be expected until the shipment leaves the port area. It has also been assumed that all foreign research reactor spent nuclear fuel would be shipped in containers, regardless of whether transport occurs via container or general cargo vessels.

Once a shipment arrives in port, the spent nuclear fuel packages would be inspected by customs officials, U.S. Coast Guard personnel, port officials, etc. Up to six inspections performed by Federal, State, and local agencies, and the shipping agent are assumed to occur for each cask shipment. The durations of these inspections are provided in Table D-11. The assumption is made that the container is opened only for the inspection conducted when the cask is first off-loaded from the vessel.

In addition to the personnel involved in the inspections, there are other port workers (longshoremen, port officials, security personnel, etc.) who would be directly involved in or co-located near the off-loading of the container, its securing to the tractor-trailer, and in the movement of the container to a staging area. (The incident-free impact of offloading operations on the ship's crew were addressed in the marine impact analysis presented in Appendix C). While arrangements are expected to be made for the immediate departure of the spent nuclear fuel from the port of entry, it is recognized that situations could occur where there may be some delay in departing the port. For example, these delays could be caused by weather or road conditions. A delay of up to 24 hours is assumed for all shipments. To account for the impact of these delays, the dose to workers not directly involved in offloading activities was estimated. In addition to workers identified in Tables D-8 through D-9, it was assumed that 50 workers are exposed to the cask for 8 hours at a distance of 50 m (163 ft). This provides a dose estimate for the 24-hour storage period.

These dose estimates are independent of port location or type. Two types of cargo vessels have been addressed in the analysis, encompassing the range of times required for offloading activities. Container vessels required the least amount of time to offload; breakbulk vessels the longest. It has been assumed that offloading operations for both containerized breakbulk cargo and container cargo at all potential ports of entry is similar. These estimates are intended to bound the potential doses associated with port activities. As discussed above, breakbulk transport of the containerized fuel casks are expected to result in the largest dose to workers due to port operations due to the longer times associated with activities that bring workers into proximity of the casks.

External radiation for an intact shipping package must be below specified limits that control the exposure of the handling personnel and general public. These limits are set forth in 49 CFR 173.

The limit of interest established therein is a limit of 10 mrem per hour at any point 2 m (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle. This limit is associated with an "exclusive-use" shipment, that is one in which no other cargo is loaded in the container used for the spent fuel transportation casks, not that the ship is an exclusive use vessel. All shipments within this program would be expected to fall within this category. In general, much of the foreign research reactor spent nuclear fuel potentially to be received would have cooled for a significant amount of time prior to shipment, resulting in external dose rates much less than the regulatory limit. Shipments of research reactor fuel in the past have had doses averaging approximately 2.3 mrem per hour at 1 m (3.3 ft) from the cask surface (see Section F.5 of Appendix F). Due to the scope of this program and the possibility that some of the fuel could be shipped fresher than has been done previously, the above cited regulatory limit has been used to estimate the worker exposures for all shipments. Appendix F, Section F.5, provides exposure rate versus distance for a transportation cask that is loaded with spent fuel that results in a dose rate at 2 m (6.6 ft) of 10 mrem per hour. This relationship was used to assign dose rates for the port activities.

Table D-12 and D-13 describe the types and numbers of personnel involved in the port activities associated with offloading the spent nuclear fuel. The times, distances, and maximum doses associated with these activities are listed for each type of personnel (all doses are simply the product of the dose rate to which the worker is exposed, based upon distance from the transportation cask, and the time the worker is exposed to this dose rate). The total port worker population and the maximally exposed individual doses are also provided. During incident-free port operations, the highest individual exposure would be to handlers and inspectors of the casks. Exposures are port-independent since it is assumed that operations would be similar at any of the potential or alternative ports of entry.

Table D-12 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Breakbulk Vessels

Exposed Workers	Exposure Distance (m)	Dose Rate (mrem/hr)	Exposure Time (minutes/cask)	Dose/Person/ Cask (mrem)	Exposed Workers	Collective Dose (Person-rem)	Individual Risk (LCF)	Collective Risk (LCF)
Longshoreman A1	0.50	37[b]	0.25	0.15	2	0.00031	6.2E-08	1.2E-07
Longshoreman A2	0.50	37[b]	3.3	2.0	2	0.0040	8.0E-07	1.6E-06
Longshoreman A3	6.00	6.4[b]	0.25	0.027	1	0.000027	1.1E-08	1.1E-08
Longshoreman B1	0.50	34	1.0	0.57	4	0.0023	2.3E-07	9.1E-07
Maximum				2.0[a]			8.0E-07a	
Subtotal						0.0066		2.6E-06
Crane Operator 1	9.00	1.8	3.0	0.090	1	0.00009	3.6E-08	3.6E-08
Maximum				0.090[a]			3.6E-08[a]	
Subtotal						0.00009		3.6E-08
Truck Driver	3.00	7.1	3.0	0.36	1	0.00036	1.4E-07	1.4E-07
Maximum				0.36[a]			1.4E-07[a]	
Subtotal						0.00036		1.4E-07
Observers	6.00	3.2	0.25	0.013	4	0.000053	5.3E-09	2.1E-08
Observers	50	0.01	480	0.0802	50	0.0040	3.2E-08	1.6E-06
Maximum				0.080[a]			3.2E-08[a]	
Subtotal						0.0041		1.6E-06
USCG Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
DOT Inspector	1.5	15	15	3.8	1	0.0038	1.5E-06	1.5E-06
NRC Inspector	1.5	15	10	2.5	1	0.0025	1.0E-06	1.0E-06
State Inspector	1.5	15	15	3.8	1	0.0038	1.5E-06	1.5E-06
Local/Port Inspector	1.5	15	5	1.3	1	0.0013	5.0E-07	5.0E-07
Receiver	1.5	15	5	1.3	1	0.0013	5.0E-07	5.0E-07
Maximum				3.8[a]			1.5E-06[a]	
Subtotal						0.013		5.2E-06
Maximum				3.8[a]			1.5E-06[a]	
Total						**0.024**		**9.6E-06**

[a] *Maximum individual exposure/risk.*

[b] *Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

Table D-12 was developed using the information pertaining to the offloading of containerized foreign research reactor spent nuclear fuel from a breakbulk vessel. The exposure times and the distances from the transportation cask used to develop the dose estimates were derived from Table D-8 and assuming the longer transfer times associated with jib or boom cranes. The exposures (worker doses) resulting from the offloading activities associated with this type of vessel are the highest, on a per cask basis, of the three types of vessels considered for transport of the foreign research reactor spent nuclear fuel: breakbulk, container, and roll-on/roll-off (the chartered or purpose-built ship could conceivably be of any of these designs). Therefore, the dose estimates derived from this data provide the upper limit to the doses that could be calculated for the offloading activities.

Alternatively, the worker doses resulting from the offloading of a foreign research reactor spent nuclear fuel cask from a container vessel result in the lowest doses per cask of the types of vessels considered for use in the shipment of the foreign research reactor spent nuclear fuel. Table D-13 was developed using the exposure times and the distances from the transportation cask developed for a container vessel which are provided in Table D-8.

Table D-13 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Containerized Vessels

Exposed Workers	Exposure Distance (m)	Dose Rate (mrem/hr)	Exposure Time (minutes/cask)	Dose/Person/Cask (mrem)	Exposed Workers	Collective Dose (Person-rem)	Individual Risk (LCF)	Collective Risk (LCF)
Longshoreman A1	0.50	37[a]	0.25	0.15	3	0.00046	6.2E-08	1.9E-07
Longshoreman A2	0.50	37[a]	0.75	0.46	1	0.00046	1.9E-07	1.9E-07
Longshoreman A3	6.00	6.4[a]	0.25	0.027	1	0.000027	1.1E-08	1.1E-08
Longshoreman B1	0.50	34	0.25	0.14	4	0.00057	5.7E-08	2.3E-07
Maximum				0.46[a]			1.9E-07[a]	
Subtotal						0.0015		6.1E-07
Crane Operator 1	6.00	32	0.50	0.027	1	0.000027	1.1E-08	1.1E-08
Crane Operator 2	6.00	32	1.6	0.085	1	0.000085	3.4E-08	3.4E-08
Maximum				0.085[a]			3.4E-08a	
Subtotal						0.00011		4.5E-08
Truck Driver	3.00	7.1	3.3	0.38	1	0.00038	1.5E-07	1.5E-07
Maximum				0.38[a]			1.5E-07[a]	
Subtotal						0.00038		1.5E-07
Observers	6.00	3.2	0.5	0.027	4	0.00011	1.1E-08	4.3E-08
Observers	50	0.01	480	0.080	50	0.0040	3.2E-08	1.6E-06
Maximum				0.080[a]			3.2E-08[a]	
Subtotal						0.0041		1.6E-06
USCG Inspector	1.5	15	5.0	1.3	1	0.0013	5.0E-07	5.0E-07
DOT Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
NRC Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
State Inspector	1.5	15	2.0	0.5	1	0.00050	2.0E-07	2.0E-07
Local/Port Inspector	1.5	15	5.0	1.3	1	0.0013	5.0E-07	5.0E-07
Receiver	1.5	15	5.0	1.3	1	0.0013	5.0E-07	2.0E-07
Maximum				1.3[a]			5.0E-07[a]	
Subtotal						0.0053		2.1E-06
Maximum				1.3[a]			5.0E-07[a]	
Total						**0.011**		**4.6E-06**

[a]*Maximum individual exposure/risk.*

[b]*Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

In both of these cases it was assumed that two transportation casks were being shipped on a single vessel and the two casks were both in the same hold. By making this assumption, the dose to the workers in the ship's hold is the result of exposure to two radiation fields during the offloading of the first casks. The impact of the presence of the second transportation cask has been included in the dose rates for the longshoremen who are in the ship's hold during the offloading activity. To simplify the analysis, it has been assumed that the dose rates for the offloading of the two casks are the same (i.e., even though when the second cask is being offloaded there is only one transportation cask in the hold, the exposures are calculated assuming that there are two casks in the hold). The total number of transportation casks shipped on a single vessel would not impact the results of this analysis. The per shipment results are for the shipment of a single cask, assuming two casks per hold. Annual exposures and exposures for the entire program do not depend on the number of transportation casks per shipment. Under the assumption that a vessel carrying more than two casks would be loaded two casks per hold, these results are solely dependent on the number of cask shipments per year and the total number of cask shipments.

There is approximately a factor of two difference between the total worker dose resulting from the use of a breakbulk vessel and the use of a container vessel per transportation cask. There is a larger difference between the dose to the maximally exposed individual (MEI). The MEI for the breakbulk vessel receives a dose of 3.8 mrem per transportation cask offloading while for the offloading of a transportation cask from a container vessel the MEI receives a dose of 1.3 mrem.

Another consideration that could affect the total worker exposure is the possibility that the vessel transporting the foreign research reactor spent nuclear fuel could make intermediate port calls between the foreign port at which the transportation cask is loaded and the port of entry for the foreign research reactor spent nuclear fuel. At the intermediate ports of call, it is possible that cargo being shipped on the vessel and in the same hold as the transportation casks could be loaded/offloaded or moved. The analysis was expanded to consider the impacts on port workers at these intermediate ports. Table D-14 provides the information used to estimate the dose to the port workers in intermediate ports. The estimates consider that the hold in which the transportation casks are being stowed have been fully loaded and that all of the cargo in the vicinity of the transportation casks must be moved at one of the intermediate ports of call. The vessel assumed in the intermediate port analysis was a breakbulk vessel. As in the analysis of the impact of the offloading of the transportation casks, this assumption results in calculations based on the type of vessel that will result in the largest estimated impact on the port workers.

Table D-14 Port Worker Exposure - Each Intermediate Port

Exposed Workers	*Distance (m)*	*Dose Rate[a] (mrem/hr)*	*Exposure Time (minutes)*	*Dose/Person (mrem)*	*Number of Workers[b]*	*Collective Dose (person-rem)*	*Individual Risk (LCF)*	*Risk per Port Call (LCF)*
Longshoreman	1.5	18	5	1.5	4	--	--	--
	5	6.4	6	0.64	4	--	--	--
	8	4.6	1	0.08	4	--	--	--
Total				**2.2**	**4**	**0.0089**	**0.00000089**	**0.0000035**

[a]*The dose rate includes the dose rate from two casks stored in the same hold.*

[b]*The same four workers are assumed to receive the entire dose from cargo handling activities in each intermediate port stop.*

The per shipment data provided in Tables D-12 through D-14 was used to develop estimates of the incident-free impact of the marine shipment of 721 transportation casks on port workers. (The number of shipments required is derived in Appendix B. The 721 shipments used in this portion of the analysis exclude all shipments of Canadian origin which are expected to be overland shipments). Table D-15 provides the results of this analysis. Data is provided for two possible shipment conditions. In the first a breakbulk vessel is used to transport all of the foreign research reactor spent nuclear fuel and this vessel is assumed to make two intermediate port calls on every voyage. During these intermediate port calls the cargo in the same hold as the transportation casks is assumed to be moved (loaded and/or offloaded) twice. The impact on port workers, in terms of population exposure and risk, in the intermediate ports is therefore twice the impact presented in Table D-14. The second set of assumptions used is that all shipments are made on a container vessel that does not make intermediate port calls. These assumptions result in a lower estimate of port worker risk since the impact of intermediate port calls is eliminated and the offloading activities for a container vessel result in lower overall doses to the port workers. These two sets of assumptions, therefore, provide estimates of the range of potential impacts on port workers.

In calculating the MEI, it was necessary to estimate the number of shipments to which a single worker could be exposed. Using the information in Table C-1, the shipments of foreign research reactor spent nuclear fuel were divided into eastern and western shipments. The eastern shipments are those that would

Table D-15 Integrated Port Worker Dose for the Basic Implementation of Management Alternative 1

	Breakbulk Vessel with 2 Intermediate Port Calls				*Container Vessel - No Intermediate Port Calls*			
	Maximally Exposed Individual (rem)	*Collective Dose to Workers (person-rem)*	*MEI Risk (LCF)*	*Worker Risk LCF)*	*Maximally Exposed Individual (rem)*	*Collective Dose to Workers (person-rem)*	*MEI Risk (LCF)*	*Worker Risk (LCF)*
Inspectors	2.0[a]	9.4	0.00080	0.0037	0.67	3.8	0.00027	0.0015
Port Handlers - Intermediate Ports	1.2	13	0.00047	0.0051	----	----	----	----
Port Handlers - Port of Entry	1.1	4.8	0.00043	0.009	0.25	1.1	0.00010	0.00044
Port Staging Personnel	0.19	3.2	0.000076	0.0013	0.21	3.3	0.00008	0.0013
Total	----	**30.2**	----	**0.012**	----	**8.2**	----	**0.0033**
Maximum	**2.0[a]**	----	**0.00080**	----	**0.67**	----	**0.00027**	----

[a]*This dose is above the allowed limit of 100 mrem/yr for the general population and would be mitigated to below the limit.*

be expected to be shipped to a port on the East Coast of the United States if the shortest shipping distance were used. Western shipments are those that would be shipped to the West Coast port. From Table C-1, 535 shipments would be considered East Coast shipments; 186 West Coast. In determining the MEI, it was assumed that all of these East Coast shipments were made through the same port, and the same workers were involved in the offloading of the transportation casks for all shipments.

The total impact on the worker population was determined by using the full 721 transportation cask shipments. Both the MEI and the collective dose to the workers have been converted into a risk estimate of LCF resulting from the doses received in offloading the transportation casks loaded with foreign research reactor spent nuclear fuel. The range of impacts for the program is from 8.2 person-rem (0.0033 LCF) (for the use of container vessels with no intermediate port calls) to 30 person-rem (0.012 LCF) (for the use of breakbulk vessels with two intermediate port calls). These risks imply that there is between a three-in-a-thousand and a one-in-a-hundred chance that this program will result in one LCF as a result of the incident-free impact on port workers. The relationship between worker dose and cancer fatalities is that 1 rem is equivalent to 0.0004 LCF.

Under the basic implementation of Management Alternative 1, shipments would be received over a 13-year period, the 10-year period for spent nuclear fuel generation plus 3 additional years to allow for the coordination of available storage, transportation casks, shipping arrangements, etc. Assuming that the shipments were evenly distributed over the 13-year period, the doses to the MEI could be in excess of the DOE and NRC limits for doses to the general public (100 mrem per year). If breakbulk vessels were used, the MEI would receive approximately 150 mrem per year on average, if no mitigation steps were taken. If container vessels were used, no individuals are expected to receive a dose in excess of the public dose limits.

The above calculations were all performed assuming that every transportation cask was shipped with an external dose rate at the selected exclusive use regulatory limit of 10 mrem hour at 2 m (6.6 ft) from the surface of the container. This provides an estimate of the upper limit to what the incident-free impacts of the offloading of the transportation casks could be. To determine a more realistic estimate of these impacts, the analysis was redone using historical data on the external dose rates associated with the transportation of research reactor spent nuclear fuel. This analysis results in an average dose rate of approximately 2.3 mrem per hour at 1 m (3.3 ft) from the cask surface, which is equivalent to a dose rate of 1 mrem per hour at 2 m (6.6 ft) from the cask surface. If the added distance from the cask surface to the container surface is not credited, this dose rate is one-tenth of the dose rate derived from the "exclusive use" regulatory limit. (See Appendix F, Section F.5)

Tables D-16 through D-19 provide the results of this analysis. No other assumptions were modified between this analysis from those used to develop the data presented earlier in this section. All of the results using the "historical" data are an order-of-magnitude lower than results derived from the use of the regulatory limit dose rates.

Table D-16 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Breakbulk Vessels (Historical Data)

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/ Person/Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	3.7[b]	0.25	0.015	2	3.1E-05	6.2E-09	1.2E-08
Longshoreman A2	0.50	3.7[b]	3.3	0.20	2	4.0E-04	8.0E-08	1.6E-07
Longshoreman A3	6.00	0.64[b]	0.25	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Longshoreman B1	0.50	34	1.0	0.057	4	2.3E-04	2.3E-08	9.1E-08
Maximum				0.20[a]			8.0E-08a	
Subtotal						6.6E-04		2.6E-07
Crane Operator 1	9.00	0.18	3.0	0.009	1	9.0E-06	3.6E-09	3.6E-09
Maximum				0.009[a]			3.6 E-09[a]	
Subtotal						9.0E-06		3.6E-09
Truck Driver	3.00	0.71	3.0	0.036	1	3.6E-05	1.4E-08	1.4E-08
Maximum				0.036[a]			1.4E-08[a]	
Subtotal						3.6E-05		1.4E-08
Observers	6.00	0.32	0.25	0.0013	4	5.3E-06	5.3E-10	2.1E-09
Observers	50	0.001	480	0.008	50	4.0E-04	3.2E-09	1.6E-07
Maximum				0.008[a]			3.2E-09[a]	
Subtotal						4.1E-04		1.6E-07
USCG Inspector	1.5	1.5	2.0	0.05	1	5.0E-05	2.0E-08	2.0E-08
DOT Inspector	1.5	1.5	15	0.38	1	3.8E-04	1.5E-07	1.5E-07
NRC Inspector	1.5	1.5	10	0.25	1	2.5E-04	1.0E-07	1.0E-07
State Inspector	1.5	1.5	15	0.38	1	3.8E-04	1 .5E-07	1.5E-07
Local/Port Inspector	1.5	1.5	5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Receiver	1.5	1.5	5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Maximum				0.38			1.5E-07	
Subtotal						1.3E-03		5.2E-07
Maximum				0.38[a]			1.5E-07[a]	
Total						**2.4E-03**		**9.6E-07**

[a]*Maximum individual exposure/risk.*

[b]*Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

The total population dose (dose to the port workers) ranges from 3.0 person-rem (breakbulk vessel with two intermediate port calls) and 0.7 person-rem (container vessel with no intermediate port calls). This corresponds to a risk of 0.0012 to 0.00033 LCF, that is, a one-in-a-thousand to a one-in-three thousand chance of incurring one LCF. For a population of workers, the relationship between exposure and LCF is

Table D-17 Port Worker Consequences from Shipment of Foreign Research Reactor Spent Nuclear Fuel on Containerized Vessels (Historical Data)

Exposed Workers	*Exposure Distance (m)*	*Dose Rate (mrem/hr)*	*Exposure Time (minutes/cask)*	*Dose/Person/ Cask (mrem)*	*Exposed Workers*	*Collective Dose (Person-rem)*	*Individual Risk (LCF)*	*Collective Risk (LCF)*
Longshoreman A1	0.50	3.7[a]	0.25	0.015	3	4.6E-05	6.2E-09	1.9E-08
Longshoreman A2	0.50	3.7[b]	0.75	0.046	1	4.6E-05	1.9E-08	1.9E-08
Longshoreman A3	6.00	0.64[b]	0.25	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Longshoreman B1	0.50	340	0.25	0.014	4	5.7E-05	5.7E-08	2.3E-08
Maximum				0.046[a]			1.9E-08[a]	
Subtotal						1.5E-04		6.1E-08
Crane Operator 1	6.00	0.32	0.5	0.0027	1	2.7E-06	1.1E-09	1.1E-09
Crane Operator 2	6.00	0.32	1.6	0.0085	1	8.5E-06	3.4E-09	3.4E-09
Maximum				0.0085[a]			3.4E-09[a]	
Subtotal						1.1E-05		4.5E-09
Truck Driver	3.00	0.71	3.3	0.038	1	3.8E-05	1.5E-08	1.5E-08
Maximum				0.038[a]			1.5E-08[a]	
Subtotal						3.8E-05		1.5E-08
Observers	6.00	0.32	0.5	0.0027	4	1.1E-05	1.1E-09	4.3E-09
Observers	50	0.001	480	0.0080	50	4.0E-04	3.2E-09	1.6E-07
Maximum				0.0080[a]			3.2E-09[a]	
Subtotal						4.1E-04		1.6E-07
USCG Inspector	1.5	1.5	0.5	0.13	1	1.3E-04	5.0E-08	5.0E-08
DOT Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
NRC Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
State Inspector	1.5	1.5	0.2	0.050	1	5.0E-05	2.0E-08	2.0E-08
Local/Port Inspector	1.5	1.5	0.5	0.013	1	1.3E-04	5.0E-08	5.0E-08
Receiver	1.5	1.5	0.5	0.13	1	1.3E-04	5.0E-08	5.0E-08
Maximum				0.13[a]			5.0E-08[a]	
Subtotal						5.3E-04		2.1E-07
Maximum				0.13[a]			5.0E-08[a]	
Total						**1.1E-03**		**4.5E-07**

[a]*Maximum individual exposure/risk.*

[b]*Includes dose from second cask in hold.*

USCG = U.S. Coast Guard, DOT = Department of Transportation

Table D-18 Port Worker Exposure - Intermediate Ports (Historical Cask External Dose Rate Data)

Exposed Workers	*Distance (m)*	*Dose Rate[a] (mrem/hr)*	*Exposure Time (minutes)*	*Dose/Person (mrem)*	*Number of Workers[b]*	*Collective Dose (person-rem)*	*Individual Risk (LCF)*	*Risk per Port Call (LCF)*
Longshoreman	1.5	1.8	5	0.15	4	--	--	--
	5	0.6	6	0.06	4			--
	8	0.5	1	0.01	4	--	--	--
Total				**0.22**	**4**	**0.00089**	**0.000000089**	**0.00000035**

[a]*The dose rate includes the dose rate from two casks stored in the same hold.*

[b]*The same four workers are assumed to receive the entire dose from cargo handling activities in each intermediate port stop.*

Table D-19 Integrated Port Worker Dose for the Basic Implementation of Management Alternative 1 (Historical Cask Dose Rates)

	Breakbulk Vessel with 2 Intermediate Port Calls				Container Vessel - No Intermediate Port Calls			
	Maximally Exposed Individual (rem)	Collective Dose (person-rem)	MEI Risk (LCF)	Risk (LCF)	Maximally Exposed Individual (rem)	Collective Dose (person-rem)	MEI Risk (LCF)	Risk (LCF)
Inspectors	0.20	0.94	0.00008	0.00037	0.07	0.38	0.00002	0.00015
Port Handlers - Intermediate Ports	0.12	1.3	0.000047	0.00051	----	----	----	----
Port Handlers - Port of Entry	0.11	0.5	0.000043	0.00019	0.03	0.11	0.000010	0.000044
Port Staging Personnel	0.02	0.3	0.000008	0.00013	0.02	0.33	0.000009	0.00013
Maximum	0.20[a]		0.00008[a]		0.07[a]		0.000027[a]	
Total		**3.0**		**0.0012**		**0.8**		**0.00033**

[a]*Maximally exposed individual.*

1 rem is equivalent to 0.0004 LCF. The MEI would receive a dose of 0.2 rem over the 13-year period of the basic implementation of Management Alternative 1. This is approximately 15 mrem per year, which is well below the NRC and DOE limits for exposure to the public (100 mrem per year).

The results of these analyses indicate that some of the port personnel that handle and inspect foreign research reactor spent nuclear fuel shipping containers could receive doses that exceed public exposure limits established by DOE and the NRC, especially when the dose rate from the casks are assumed to be at the regulatory limit for exclusive use shipments of 10 mrem per hour measured 2 m (6.6 ft) from the surface of the shipping container. The analyses results are conservative due to three factors. First, it is estimated that for most shipments the external dose rate for the loaded transportation cask would be near the historic dose rates, which average a factor of ten below the regulatory limit. Second, the analyses assumed that the same port inspectors and handlers handle all shipments. In reality, most port personnel work on shifts, so the likelihood of all shipments being handled by the same shift is low. Finally, all of the shipments passing through any East Coast port were assumed to pass through the same port. In reality, it is more than likely that the shipments would be made through more than a single port.

However, the existence of some shipments with external dose rates closer to the exclusive use regulatory limit suggests that DOE should provide a means to assure that individual port personnel do not receive doses in excess of the public dose limits. As a minimum, the program should establish administrative procedures that would maintain records of the exposure rates associated with each shipment and the ports of departure and entry. The measurement of interest for the record keeping would be the external dose rates outside the container, which houses the transportation cask since the port personnel do not enter the container. These measurements could be used to identify shipments that would result in port personnel exposures above those calculated based on the historical spent nuclear fuel transportation external dose rate. By tracking this information, DOE would be able to identify if and when additional precautions to reduce individual exposures should be taken.

D.4.6 Cumulative Port Impact Analysis Methodology

Analyses have been carried out to estimate the maximum occupational doses associated with the port activities segment of the transportation of foreign research reactor spent nuclear fuel. Since port workers are expected to be exposed to other shipments of radioactive materials, the cumulative impact of all radioactive material shipments has been estimated. The cumulative analysis is necessary to determine the impact on port workers from doses received through actions associated with the foreign research reactor spent fuel return program and through other actions, both DOE and commercially initiated.

The maximum exposure for a worker involved in transporting the foreign research reactor fuel is predicted to result from activities associated with the unloading of the spent fuel casks in port, cask inspection, and cask preparation for truck shipment to the management sites. If the same individuals were present for all proposed shipments of foreign research spent nuclear fuel on an annual basis (a conservative assumption), the maximum dose would be approximately 150 mrem, as discussed in the previous section. This estimate is based on the use of the "exclusive use" regulatory external dose rate. Based on historical spent nuclear fuel shipment data, this maximum annual dose would be 15 mrem.

Since commercial ports routinely receive other shipments of radioactive materials under other DOE programs or other commercial activities, the port worker would also be potentially exposed to additional sources of radiation. To estimate the annual exposure rate of port workers resulting from handling of commercial radioactive material shipments, the following must be determined.

- Number of radioactive packages handled per year
- Length of exposure time per package
- Dose rate per package

Records of shipments through the potential ports of entry were used to estimate the annual throughput of packages with radioactive contents. Radioactive materials were identified by the product code listed for each shipment. The radioactive shipments were then grouped into six categories and exposure rates at 1 m (3.3 ft) from the outer surface of the package were assigned for each group as follows:

• enriched uranium hexafluoride	(0.5 mrem per hour)
• normal uranium hexafluoride	(0.2 mrem per hour)
• depleted uranium	(0.2 mrem per hour)
• uranium oxide	(0.2 mrem per hour)
• spent nuclear fuel (foreign research reactor)	10 mrem per hour [at 2 m (6.6 ft) from the container surface]
• other radioactive materials	(0.2 mrem per hour)

Each shipment record lists the weight and number of packages included in the shipment. Since package descriptions were not uniform and included units, containers, cases, boxes, barrels, drums, packages, cartons, etc., the assumption was made that the radioactive shipments would be stacked on skids and the total number of skids per shipment, rather than the number of packages per shipment, would be used to estimate the dose received by workers. The weight and number of individual shipments was examined for each shipment to estimate the number of skids. In most cases, boxes, cartons, barrels, and drums were assumed to be handled four to a skid. When a large number of light packages was included in one shipment, these were assumed to be handled as either eight or 32 packages per skid.

The annual dose to port workers resulting from handling commercial radioactive shipments were estimated based on the number of shipments passing through the port and an estimated handling time of ten minutes per skid or cylinder. Each port typically uses three shifts per day and therefore workers were assumed to be exposed to one-third of the packages passing through the port. This is a conservative assumption given that there are typically many berths and terminals within one port, thus making it unlikely that one individual would be present for even one-third of the shipments of radioactive materials. The estimated dose to the MEI from these commercial shipments is shown in Table D-20.

Table D-20 Estimated Maximum Exposure to Dock Workers from Commercial Shipments of Radioactive Material

Port	*Average No. of Radioactive Shipments per Year*	*Estimated Maximum Exposure per Year (mrem)*	*Port*	*Average No. of Radioactive Shipments per Year*	*Estimated Maximum Exposure per Year (mrem)*
Baltimore, MD	31	3.4	New Orleans, LA	7	3.9
Boston, MA	2	0.2	Norfolk, VA	30	3.9
Charleston, SC	16	3.1	New York, NY	104	16.8
Fernandina Beach, FL	21	less than 0.1	Oakland, CA	39	9.0
Galveston, TX	1	less than 0.1	Philadelphia, PA	1	less than 0.1
Houston, TX	14	4.0	Portland, OR	1	0.6
Jacksonville, FL	4	0.3	Portsmouth, VA	28	5.5
Long Beach, CA	1	less than 0.1	Port Everglades, FL	7	0.1
Los Angeles, CA	6	0.2	Savannah, GA	7	1.5
Miami, FL	1	less than 0.1	Wilmington, NC	2	1.2

As this table shows, yearly exposures for the commercial shipments are typically less than 10 mrem per year, which is well within the regulatory limit of 100 mrem per year established for a member of the general public. New York (at 16.8 mrem per year), which had the most commercial shipments of radioactive material on a yearly basis, was the only port to exceed 10 mrem per year. However, the Port of New York consists of three terminals in Elizabeth (NJ), Brooklyn, and Manhattan. This diversity means that in practice, the average port worker would be involved in only a portion of the shipments through "New York."

Some of the potential ports are being used or have the potential to be used for other DOE-initiated activities. These activities include the purchase of Russian low enriched uranium (LEU) under the agreement Suspending the Antidumping Investigation of Uranium from the Russian Federation and the import of Russian LEU derived from the dismantling of nuclear weapons in Russia. Estimated maximum exposures from these activities are 0.9 mrem and 1.4 mrem per year, respectively.

The impact of all of these shipments can be viewed in two ways. If the foreign research reactor spent nuclear fuel shipments were to have dose rates like the historical data indicate they would, the total maximum worker exposure from all of these activities would be well below the public dose limits (by at least a factor of three). If the foreign research reactor spent nuclear fuel shipments were to be closer to the external dose rate allowed by the "exclusive use" regulatory limit, these other activities do not significantly alter the maximum worker dose. In this case, DOE's response to the worker exposure would be dictated by the exposure resulting from the shipment of foreign research reactor spent nuclear fuel.

D.4.7 Incident-Free Port Impacts of Alternatives to the Basic Implementation of Management Alternative 1

Three alternatives to the basic implementation of Management Alternative 1 were identified that could impact the incident-free port risk calculations that were performed. (Chapter 2 describes the alternatives to the basic implementation of Management Alternative 1.) The implementation subalternative of *accepting spent nuclear fuel only from developing countries*, which are identified as countries other than high-income economies, would result in a reduction in the amount of spent nuclear fuel transported by ship. Table C-12 listed the countries that are considered to be countries other than high-income economies and the number of foreign research reactor spent nuclear fuel shipments that would be required to transport their spent nuclear fuel to the United States. One hundred sixty-eight transportation casks would be shipped to the United States under this implementation subalternative. Under the *foreign research reactor spent nuclear fuel for 5-years only* implementation subalternative, the number of shipments of foreign research reactor spent nuclear fuel would be reduced to 586 shipments requiring ocean transport. (The derivation of the number of shipments required in this alternative is presented in Appendix B.)

The third alternative, with the capability to impact the results of the incident-free port risk analysis, is the *overseas processing of the foreign research reactor spent nuclear fuel with the shipment of the vitrified waste to a storage facility in the United States*. Under this alternative, eight transportation cask shipments of vitrified waste could be made.

In addition to these alternatives, a hybrid alternative was analyzed. In this alternative, those countries that have the capability to store high-level waste would be encouraged to process the aluminum-based research reactor spent nuclear fuel and to accept for storage the resulting high-level waste. (For this alternative these countries are assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom). The United States would accept for storage the foreign research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability. In this alternative, this includes all of the countries identified in Table C-1, except for those listed above. Under this hybrid alternative, 452 shipments of foreign research reactor spent nuclear fuel are assumed to be sent to the United States, excluding overland shipments of Canadian origin.

The incident-free port risks associated with these three alternatives are discussed in the following sections.

Implementation Subalternative 1a of Management Alternative 1 – Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only From Developing Countries: Developing countries are defined as countries other than high-income economies. As stated above, this implementation subalternative would result in the shipment of 168 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the basic implementation of Management Alternative 1 incident-free port impact have been used in the analysis of this subalternative. To compare this subalternative to the basic implementation of Management Alternative 1, it is only necessary to perform the analysis using one external dose rate, either the regulatory dose limit or the historic dose rate. The regulatory dose rate was chosen for the comparison.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are considered for the breakbulk vessel but not for the container vessel.

- Two transportation casks are being transported in the same hold on each cargo vessel.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per shipment information change. The 168 shipments required to meet the needs of this subalternative would result in a reduction in the total (program) impacts by approximately 77 percent. The total population exposure would range from 7.0 person-rem (for the breakbulk vessel with two intermediate port calls) to 1.9 person-rem (for the container vessel with no intermediate port stops). This corresponds to an incident-free risk of 0.0028 to 0.00076 LCFs (i.e., a chance of between three-in-a-thousand and seven-in-ten thousand of incurring one LCF).

Implementation Subalternative 2a of Management Alternative 1 – Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5 Year Policy Duration: As stated above, this implementation subalternative would result in the shipment of 586 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used previously for incident-free port impact have been used in the analysis of this subalternative. This implementation subalternative has been analyzed using the "exclusive use" regulatory limit transportation cask external dose rates. To compare this implementation subalternative to the basic implementation of Management Alternative 1, it is only necessary to perform the analysis using one external dose rate.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are considered for the breakbulk vessel but not for the container vessel.
- Two transportation casks are being shipped in the same hold of each cargo vessel.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. Therefore, none of the assumptions used to generate the per shipment information change. The 586 shipments required to meet the needs of this implementation subalternative would result in a reduction in the total (program) impacts to approximately 81 percent of the impacts associated with the basic implementation of Management Alternative 1. The total population exposure would be 25 person-rem (for the breakbulk vessel with two intermediate port calls) to 6.7 person-rem (for the container vessel with no intermediate port stops). This corresponds to an incident-free risk of 0.0098 to 0.0027 LCFs (i.e., a chance of between one-in-a-hundred and three-in-a-thousand of incurring one LCF).

Management Alternative 2, Subalternative 1b – Overseas Reprocessing with Shipment of the Vitrified Waste to a U.S. Storage Facility: In this subalternative under Management Alternative 2, the foreign research reactor spent nuclear fuel would be processed overseas (most probably in Great Britain or France) and the waste products are contained within several vitrified waste logs. This high-level waste may be brought to the United States for storage in one of the storage facilities evaluated under this EIS. Under these conditions, up to eight transportation casks containing vitrified waste would be shipped from Europe to the United States. This analysis addresses the incident-free port risks associated with transporting these eight casks of vitrified waste from Europe to the United States.

As with the shipment of foreign research reactor spent nuclear fuel as spent nuclear fuel, the primary incident-free port impacts of shipping vitrified waste would be upon the workers in the ports. The assumptions used in the analysis of the incident-free port impact of the basic implementation of Management Alternative 1 have been used in the analysis of this subalternative. Differences between the foreign research reactor spent nuclear fuel transportation casks and the vitrified waste transportation casks are not expected to significantly alter the work requirements in port. For the purposes of this analysis, it has been assumed that the vitrified waste would be transported on a chartered vessel, and there would be no intermediate port calls.

This alternative has been analyzed using the regulatory limit transportation cask external dose rates. Little information is available on the casks to be used to transport the vitrified waste. No attempt was made to extrapolate limited historical data to determine the port worker incident-free impacts from any other exposure rate other than the limit set forth in NRC and DOE regulations.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask are as detailed in Tables D-8 through D-10.
- The intermediate port stops are not considered for the container vessel.
- Two transportation casks are being transported in the same hold of the cargo vessels.

The per shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per shipment information change. The eight shipments required to meet the needs of this subalternative would result in a reduction in the total (program) impacts by a factor of approximately one hundred. The total population exposure would be 0.0091 person-rem for the container vessel with no intermediate port stops. This corresponds to an incident-free risk of 0.0000036 LCFs (i.e., a chance of approximately four-in-a-million of incurring one LCF).

Hybrid Alternative – Acceptance of Foreign Research Reactor Spent Nuclear Fuel From Countries Without High-Level Waste Disposal Capability: As stated above, this hybrid alternative results in the marine shipment of 452 transportation casks of foreign research reactor spent nuclear fuel. The assumptions used in the analysis of the incident-free port impact of the basic implementation of Management Alternative 1 have been used in the analysis of this alternative. This alternative has been analyzed using external dose rates derived from the exclusive use regulatory limit for a transportation cask.

Included in the assumptions that have not changed in this analysis are the following:

- The worker exposure times and distances from the transportation cask.
- The intermediate port stops are considered for the nonchartered vessel but not for the chartered vessel.
- Two transportation casks are being shipped in the same hold of each cargo vessel.

The per-shipment incident-free impact on the port workers would be identical to that calculated for the basic implementation of Management Alternative 1. None of the assumptions used to generate the per-shipment information changes. The 452 shipments required to meet the needs of this hybrid alternative would result in a reduction in the total (program) impacts to approximately 63 percent of the impacts associated with the basic implementation of Management Alternative 1. Therefore, the total population exposure would be 19 person-rem (for regularly scheduled commercial vessel with two intermediate port calls) to 5.1 person-rem (for the chartered vessel with no intermediate port calls). This corresponds to an incident-free risk of 0.0076 to 0.0021 LCFs (i.e., a chance of between approximately one-in-five hundred to less than one-in-a-hundred of incurring one LCF).

D.5 Accident Impacts: Methods and Results

D.5.1 Introduction

This section describes the approach used to assess the risks associated with in-port accidents that could result in a release of radioactive material from the transportation cask containing foreign research reactor spent nuclear fuel. The discussion addresses both the accident risk assessment methodology and the results of the analyses. The risk assessment results are presented in terms of a per-shipment accident risk and the total port-accident risks associated with various alternative under the proposed action.

Spent nuclear fuel shipments could occur via any of four types of vessels, container ships, roll-on/roll-off vessels, breakbulk vessels, and purpose-built (dedicated) vessels. In the incident-free analysis, only breakbulk vessels and container vessels were studied, since these two provide a bounding assessment of the risks associated with port activities. Under the assumptions used in the port accident analysis, the type of vessel used to transport the foreign research reactor spent nuclear fuel would not impact the result of the analysis.

All radiologically-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent, which is the sum of the effective dose equivalent (EDE) from the external radiation exposure and the 50 year committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of person-rem for collective population and rem or mrem for individuals. The impacts are further expressed as health risks, specifically in terms of LCF. The health risk conversion factors were derived from International Commission on Radiological Protection Publication 60 (ICRP, 1991). See Chapter 4 for a more complete explanation of radiation measurement and health risks.

D.5.1.1 Accident Risks

Risk (R) is the product of the magnitude (M) of an unfavorable consequence and the probability of occurrence (P) of that consequence. Thus,

$$R = PM.$$

For accidents that happen during the transportation of radioactive materials, the unfavorable consequences of the accident may include exposure of people to radiation emitted by the radioactive materials released to the atmosphere by the accident and the occurrence of radiation induced health effects that the exposure may cause. The magnitude of these consequences depends on the amount of radioactivity released to the atmosphere, the degree to which the radioactive materials are diluted during downwind transport, and the size of the population that is exposed to radiation from the passing plume or from materials deposited on the ground or in the water from the plume. The amount of dilution experienced by a plume during downwind transport depends principally on atmospheric stability and windspeed. The size of the exposed population is determined by the direction the wind is blowing at the time of the accident and the number of people in that direction. Thus, the probability that a given consequence occurs is given by the following product,

$$P = P_{st} P_w P_p$$

where P_{st} is the probability of the source term (the amount of radioactive material released), P_w is the probability of the prevailing weather conditions, and P_p is the exposure probability of the population that is exposed to radiation, given the direction that the wind is blowing at the time of the accident.

D.5.1.2 Ship Accident Risks

The total risk caused by transporting foreign research reactor spent fuel to and within the United States is the sum of the risks for transport by land and by ship. Thus,

$$R_{total} = R_{land} + R_{ship}$$

For ships, the risk is given by:

$$R_{ship} = R_{sea} + R_{coast} + R_{port}$$

where R_{sea}, R_{coast}, and R_{port} are the risk while at sea, while sailing in coastal waters, and while in the port (R_{sea} and R_{coast} were addressed in Appendix C). Each risk term has an incident-free and an accident contribution, so

$$R_{port} = R_{port\text{-}incident\text{-}free} + R_{port\text{-}accident}$$

The accident risks associated with the foreign research reactor spent nuclear fuel while it is on a ship in the port, $R_{port\text{-}accident}$, is the subject of this section. $R_{port\text{-}incident\text{-}free}$ was covered in D.4 of this appendix.

The only port accidents considered are those where the ship carrying the spent nuclear fuel is struck by another ship. Accidents where the spent nuclear fuel transport ship rams a fixed structure (a bridge or a dock), rams another ship (a collision where the spent nuclear fuel ship is the striking ship), or runs aground are neglected for the following reasons.

First, ship accident data show that when a ship rams a fixed structure or collides with another ship, damage to the striking ship is confined to its prow and to the forwardmost hold. Even in these cases, the forces exerted on cargo in the forward hold are less than the forces exerted on cargo in the case where a striking ship impacts the cargo hold.

Second, because keel structures are massive and very sturdy, groundings rarely lead to significant damage to cargo, although monetary losses due to sinking of cargo or the ship can be significant. Immersion to the depths of harbor channels is unlikely to damage a spent nuclear fuel cask or pose a significant retrieval problem; therefore, groundings are also neglected in this study.

D.5.2 Risk Analysis Methods

The consequences of ship collisions that occur in ports were estimated using the MELCOR Accident Consequences Code System (MACCS) (Jow et al., 1990, Sprung et al., 1990), originally developed by Sandia National Laboratories and the NRC for use in estimating the consequences of nuclear power plant accidents. The MACCS code was selected for these analyses because it can model an accident that takes place at a specific location and, more importantly, can model the site-specific population distribution around that location including space that is ocean and thus unpopulated.

If a ship transporting spent nuclear fuel is struck by another ship, and the collision leads to the failure of the spent fuel cask, the prevailing winds would transport the radioactive gases and aerosols in the plume released to the atmosphere during the accident away from the accident scene. During transport by the prevailing winds, downwind populations would likely be exposed to radiation, and land, buildings, and crops located below the plume trajectory might be contaminated by the radioactive materials deposited from the plume. Estimation of the range and probability of the health effects induced by the radiation exposures, and of the economic costs and losses that would result from any contamination of land, buildings, and crops is the objective of a MACCS accident consequence analysis.

MACCS calculations require the following accident and site data:

The radioactive inventory of the cask at the time of the accident for those radionuclides important for the calculation of accident consequences.

Release fractions and probability of release for the source term caused by the accident.

Plume characteristics for the radioactivity released to the atmosphere by the accident, the sensible heat content and the release time and duration.

Meteorological data characteristic of the region where the port is located, usually one year of hourly readings of windspeed, atmospheric stability, and rainfall.

The population distribution about the port where the accident occurs.

Emergency response assumptions, such as evacuation time and average speed; building shielding factors and the time when people take shelter if nearby populations are instructed to take shelter.

Land usage (habitable land fractions and farmland fractions) for the region surrounding the port.

Given these data, MACCS predicts:

The downwind transport, dispersion, and deposition of the radioactive materials released from the failed spent fuel cask.

The radiation doses received by the exposed populations via direct (cloudshine, inhalation, groundshine, resuspension) and indirect (ingestion) exposure pathways.

The mitigation of these doses by emergency response actions (evacuation, sheltering, and post-accident relocation of people).

Health effects that might occur in the population exposed to radiation as a result of the accident, both LCF and acute injuries (if short-term exposures are large).

The potential costs of emergency response actions, and of the decontamination, temporary interdiction, and condemnation of milk, crops, land, and buildings located in the region around the port, if necessary.

D.5.3 MACCS Input Data

D.5.3.1 Source Terms

MACCS source terms are specified by five input quantities: the probability (P_{st}) of the accident that leads to the release; the time (t) and duration (delta t) of the release (for ship accidents there may be both a mechanical release following the collision and a later thermal release if the accident progression leads to a fire); and the accident release fraction (f_i) and cask inventory (I_i) of each radionuclide (i) important for the calculation of accident consequences.

D.5.3.1.1 Source Term Probabilities

In the Environmental Assessment for the Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d), accident risks were estimated using six categories of accident severity. To facilitate comparison of the risk estimates developed for this EIS to those developed for the Environmental Assessment, the EIS retained these six categories of accidents. Table D-21 presents the six categories of accident severity used in the EIS (and Environmental Assessment), the values of the conditional release probabilities (conditional on the occurrence of the specified accident), and the radionuclide release fractions used in the EIS for each severity category.

Inspection of the table shows that no radioactive releases are expected for accidents that fall into severity categories 1 or 2. Accidents that fall into category 3 fail the cask's seal but not the fuel elements contained within the cask. Thus, only radioactivity produced by activation of chemical deposits located on the outside of the fuel elements corrosion deposits can be released. Since research reactor fuel is not significantly plagued by corrosion deposit formation, corrosion deposits are negligible for research reactor spent fuel. Although the accident phenomenology specified for category 6 is more severe than that for category 5, and that for category 5 is more severe than that for category 4, in the Environmental Assessment all three of these categories were assigned a conditional probability of occurrence of 0.0004. Since increasing accident severity should mean decreasing accident probability, the conditional probabilities assigned to these categories should not be identical. Although the Environmental Assessment release fractions given in the table were retained as the base case for analysis, a method to develop new estimates of the conditional probabilities of occurrence for categories 4, 5, and 6 was formulated. That method is presented below.

D.5.3.1.2 EIS Source Term Probability Considerations

Table D-22 presents a sequence of events that encompasses the accident conditions associated with accident severity categories 4, 5, and 6. This sequence of events provides a reasonable description of a severe collision between large ships that leads both to a severe fire and to a release of radioactivity from the violated spent fuel cask.

This construct allows source term probabilities (P_{st}) to be estimated as the product of the probabilities of occurrence for the seven events. Table D-23 shows how values for P_{st} were calculated in this analysis for accident severity categories 4 through 6. $P_{collision}$ and $P_{severe\ fire}$ were estimated from ship accident data. Because data were sparse for some of the ports studied, these probabilities were not developed separately for each port (i.e., dependencies on port traffic were neglected). P_{hold} and $P_{engulfing\ fire}$ were derived from ship specifications (number of cargo holds and the dimensions of these holds for the prototypic breakbulk freighter used in the impact and crush analyses). P_{impact} and P_{crush} were estimated, as is described in

Table D-21 Accident Severity Categories Used in the EIS

Accident Severity Category	Accident Conditions	Conditional Probabilities	Radionuclide Release Fractions[a]
1	Conditions do not exceed those for a Type B package; no release of contents.	0.603	Co 0 Kr 0 Cs 0 Ru 0 Part 0
2	Conditions equal to those for Type B certification tests; no release of contents.	0.395	Co 0 Kr 0 Cs 0 Ru 0 Particulate 0
3	Seal damage creates leak path, but fuel undamaged; only corrosion deposits, if present, released from package.	0.002	Co 0.012 Kr 0 Cs 0 Ru 0 Particulate 0
4	Impact damage great enough to cause damage to spent fuel; fuel particulates and fission gases may be released.	0.0004	Co 0.012 Kr 0.010 Cs 0.00000001 Ru 0.00000001 Particulate 0.00000001
5	Impact damage to seals plus fire severe enough to cause the cask to leak with release of fission gases, volatiles, and particulates.	0.0004	Co 0.012 Kr 0.100 Cs 0.0009 Ru 0.000001 Particulate 0.00000005
6	Severe impact damage plus fire severe enough to oxidize fuel with release of greater amounts of volatiles than Category 5.	0.0004	Co 0.012 Kr 0.100 Cs 0.00098 Ru 0.000042 Particulate 0.00000005

[a] *No credit was taken for the deposition of fission product vapors or aerosols released from a failed cask onto surfaces of the ship or cargo.*

Table D-22 Event Sequence for a Severe Ship Accident

Event	Event Probability
Collision between large ships	$P_{collision}$
Foreign research reactor spent nuclear fuel hold struck	P_{hold}
Foreign research reactor spent nuclear fuel hold penetrated (the cask and fuel are subjected to impact forces)	P_{impact}
Cargo compression (the cask is subjected to crush forces)	P_{crush}
Severe fire ensues	$P_{severe\ fire}$
Fire engulfs the cask (heat loads are sufficient to vaporize cesium)	$P_{engulfing\ fire}$
Convective flow of air through cask causes ruthenium to oxidize	$P_{convection}$

Attachment D4, using the methods of Minorsky (Minorsky, 1959) and results from previous studies of ship accidents (ORI, 1981b). $P_{convection}$ was estimated by review of data on fires and on the temperatures required to oxidize ruthenium to RuO_4, which is necessary to yield the higher ruthenium release fractions.

Table D-23 EIS Source Term Probability Expressions

Accident Severity Category	*Probability*
4	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})$
5	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}$
6	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}P_{convection}$

D.5.3.1.3 Probabilities Developed From Accident Data

Fifteen years of Lloyd's casualty data (Lloyds, 1991) and previous studies of ship accidents (Warwick, 1976; SRI, 1978; ORI, 1981a; Abkowitz, 1985) were reviewed to develop (1) the probability of a severe collision ($P_{collision}$) between large ships that occurs dockside in ports or while sailing in port channels, and (2) the probability that such a collision leads to a severe fire (P_{fire}).

Collision Probability

Ship accident casualty data for the years 1978 through 1993 and U.S. port call data for the years 1992 and 1993 were obtained from Lloyd's Maritime Information Services, Inc. Searches of the port call data for the 2-year period 1992-1993 identified the number of port calls made in U.S. ports by all ships, all dry cargo ships, and all dry cargo ships of deadweight 10 to 20 thousand long tons (equivalent to approximately 10,160 to 20,321 metric tons or 11,200 to 22,400 tons). The searches were performed twice, once restricting the results to collision that occurred in port waters only and once adding collisions that occurred in restricted approaches (rivers) that lead to the port. The addition of restricted approach waters was done to permit comparison to results from the literature that included or seemed to include collisions in the river that leads to a port.

The collision frequency per port call is based on a relatively small numbers of collisions. The 15 years of Lloyd's data contained only 69 collisions that occurred in U.S. ports or the restricted river waters that lead to them. Because of this, it is inappropriate to select a value for $P_{collision}$ that is significantly more precise than an order-of-magnitude estimate. The Lloyd's data indicate that for all types of commercial vessels in all U.S. ports, the number of collisions per port call is 0.000077. Other studies provide a range of values for collisions per port call (Warwick, 1976; SRI, 1978; ORI, 1981a; Abkowitz, 1985); however, the Lloyd's database is the most inclusive and the largest (based on approximately 900,000 port calls), so the result based on their data was used here. As discussed earlier, only an order-of-magnitude value is warranted, so the 0.000077 collision per port call was rounded up to 0.0001 ($P_{collision} = 0.0001$).

Probability of Severe Fires

The sources of information cited above were examined to determine an estimate of the probability of a severe fire, given a ship collision. Four estimates of this probability were developed. The 15 years of Lloyd's casualty data contains 1,073 ship collisions in ports located anywhere in the world. Eleven of these collisions led to fires, five caused extensive fire damage, and one involved buckling of structures due to thermal loads. Therefore, the Lloyd's data suggest that the chance that a ship collision leads to a severe fire is 5/1073 = 0.0045.

Only one of the 83 collisions identified by Warwick and Anderson (Warwick, 1976) led to a fire. However, that fire consumed one of the ships involved, the *Sea Witch*. Thus, the Warwick and Anderson data suggest that the chance that a collision will lead to a severe fire is 1/83 = 0.012.

Only 17 of the 391 collisions in the Abkowitz and Galarraga study (Abkowitz, 1985) led to fires of any severity. Thus, the probability that a collision leads to a fire of any severity is 17/391 = 0.044. SRI data suggest that about five percent of all ship fires involve an entire hold (SRI, 1978). Thus, the chance that a ship fire on a cargo ship will involve an entire hold is about 0.05. Combining these last two results allows the probability that a cargo ship collision leads to a severe fire to be estimated as follows:

(fires per collision) x (fires involving an entire hold per fire) =

$$(4.4x10^{-2})\ x\ (0.05) = 0.0022 \text{ severe fires per collision}$$

Fires on cargo ships were reviewed by several countries for the International Maritime Organization. The French submission (IMO, 1992) to the International Maritime Organization developed data for 599 cargo ship fires that took place during the 11-year period 1978-1988. Only 2 of the 599 fires were caused by ship collisions. Thus, the probability that a collision leads to a fire of any severity is 2/599 = 0.017. Of the 599 fires, 122 led to immediate total loss, and 195 led to damage first thought to be repairable but which later was determined to be beyond repair. Thus, the chance that a fire is severe is greater than 122/599 = 0.20 and less than (122+195)/599 = 0.53. If the average of these two estimates is used, then the probability that a collision leads to a severe fire can again be calculated as was just done above:

(fires per collision) x (fires resulting in total loss per fire) =

$$(1.7x10^{-2})\ x\ (0.37) = 0.0063 \text{ severe fires per collision}$$

If these four estimates for severe fires per collision are averaged, a value of 0.0063 results. Rounding to the nearest order-of-magnitude suggests that $P_{severe\ fire} = 0.01$ is a reasonable estimate for the chance that a severe fire will be caused by a ship collision.

No credit is taken for fighting of hold fires during accidents that occur in U.S. ports, all of which have fire fighting equipment, even though fighting of hold fires with water should keep fire temperatures well below those assumed in this study.

D.5.3.1.4 Probability of Mechanical Loads That Cause Damage

A severe ship collision could damage a spent nuclear fuel transportation cask and the elements contained in the cask by subjecting the cask to impact forces, crush forces, and/or thermal loads. Because force is the derivative of energy with distance, both impact forces and crush forces at any penetration distance (d) can be estimated by differentiating expressions that give the dependence on distance of the kinetic energy that is dissipated during the collision. Attachment D4 provides the details of this analysis. In Section 1 of Attachment D4, the kinetic energy associated with ship collisions is discussed. Next, in Section 2, the impact forces required to damage a cask and/or the fuel elements inside the cask are estimated. The crush forces required to damage a cask or the fuel elements inside the cask are described in Section 3.

The kinetic energy associated with ship collisions has been studied (Minorsky, 1959) and extended to develop correlations between penetration depth and energy absorbed. The methodology addresses the evaluation of the kinetic energy, impact forces, and crush forces and their relationship to the impact and crush probabilities (P_{impact} and P_{crush}) associated with ship collisions. The results of this evaluation concluded that P_{crush} is equal to 0.40, and P_{impact} is equal to 0.0.

D.5.3.1.5 Probabilities Developed From Ship Design Data

Two probabilities can be derived from the general ship design data, P_{hold} and $P_{engulfing\ fire}$. The first of these probabilities addresses the likelihood that the collision results in damage to the hold in which the spent nuclear fuel cask resides. (If the cask is stowed in an aft hold and the collision results in damage to a forward hold, no cask damage would be expected.) The second probability addresses the likelihood that the severe fire resulting from the accident (see Section D.5.3.1.3) is located in the same hold and on the same deck as the cask of spent nuclear fuel.

If foreign research reactor spent fuel casks were shipped one at a time, as is assumed here, then P_{hold}, the probability that the hold that contains the cask is the hold that is struck, can be approximated by $1/N_{hold}$, where N_{hold} is the number of holds in the ship transporting the spent nuclear fuel cask. The representative breakbulk freighter used in the impact and crush analyses described below has seven holds. Therefore, for this prototypic ship, $P_{hold} = 1/7 = 0.143$.

The total cargo area of this typical breakbulk freighter is about 3,066 m^2 (33,000 ft^2). Each hold includes two, three, or four decks. Together, the seven holds encompass 21 decks. Thus, the area of each deck is about 3,066/21 = 146 m^2 (33,000/21 = 1,600 ft^2). The Pegase cask used as a prototype in this study has a 2.1-m by 3-m (7-ft by 10-ft) base. This cask should be completely engulfed by a pool fire that has a diameter of 9.1 m (30 ft), provided that the fire occurs in the same hold and on the same deck that the cask is stored on. Since a pool fire of diameter 9.1 m (30 ft) occupies about 65 m^2 (700 ft^2), any engulfing fire will probably involve an entire deck in a hold. If a collision can lead to a fire on any deck in the hold, the $P_{engulfing\ fire} = 1/21$. Limiting the location of the fire to the struck hold or an adjacent hold reduces the number of decks on which the fire could occur. In this case, the number of holds of interest is approximately ten, and therefore, $P_{engulfing\ fire} = 1/10$. Using the larger estimate gives $P_{engulfing\ fire} = 0.1$.

D.5.3.1.6 Probability of Convective Flow Through the Failed Cask

Nonuniform heating of the cask during engulfing fires is expected to produce substantial flow of gases through the cask if two or more small holes or one medium hole have been produced in the cask by the ship collision. Because transportation cask bottoms and lid seats are welded to the cylindrical shell of the cask using full-penetration welds that are as strong or stronger than the parent material, when the cask shell is subjected to a severe stress (e.g., high impact or crush forces), the cask shell should yield before the welds fail. In fact, extra-regulatory 97 km/hr (60 mph) drop tests produced large plastic strains in the cylindrical shell of the test cask without failing its welds (Ludwigsen and Ammerman, 1995). Thus, during a ship collision, crush forces should collapse the cask walls inward without producing catastrophic failure of the lid, its seat, or the welds that attach the seat or the bottom of the cask to the cask walls. Therefore, an unusual configuration of cargo and/or deformed ship structures must be produced during the ship collision in order to subject the cask to forces that will produce failures substantially worse than failure of the lid seal. Either the lid seat must be bent significantly, or at least two penetrations must break, or the cask walls must be sheared or punctured. Although data for such failures is lacking, because cask designs normally do not fail by these mechanisms, the probability that a failure substantially worse than seal failure occurs is conservatively assumed to be no larger than 0.1, therefore $P_{convection} = 0.1$.

D.5.3.1.7 Source Term Probability Values

Table D-24 summarizes the estimates developed for the probabilities that enter the EIS source term probability expressions presented in Table D-23.

Table D-24 Estimated Values for Probabilities in Source Term Probability Expressions

Severity Category	Probability	Estimated Value[a]
	$P_{collision}$	0.0001
	P_{hold}	0.143
	P_{impact}	0.0
	P_{crush}	0.40
	$P_{severe\ fire}$	0.01
	$P_{engulfing\ fire}$	0.1
	$P_{convection}$	0.1
4	$P_{st} = P_{collision}P_{hold}(P_{impact} + P_{crush})$	0.000006
5	$P_{ST} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}$	5×10^{-9}
6	$P_{ST} = P_{collision}P_{hold}(P_{impact} + P_{crush})P_{severe\ fire}P_{engulfing\ fire}\ P_{convection}$	6×10^{-10}

[a]*Severity category 6 is a subset of severity category 5, which in turn is a subset of severity category 4. Therefore, the final estimated value for each P was adjusted to account for this.*

D.5.3.1.8 Source Term Magnitudes

In MACCS, source term magnitudes (M_{sti}) are given by the product of the inventory (I_i) of radionuclides (i) available for release and the fraction (f_i) of that inventory that is released during the accident being examined. Therefore,

$$M_{sti} = I_i f_i.$$

Cask radionuclide inventories were developed for three types of research reactor fuel — Training, Research, Isotope, General Atomic (TRIGA), RHF, and BR2 — for use in the port accident analysis (see Appendix B). Table D-25 presents these inventories. Because it is partly metallic, the TRIGA fuel may undergo exothermic oxidation if exposed to air while at elevated temperatures during an accident involving an enveloping fire.

Because of the large number of casks that might be used to transport foreign research reactor spent nuclear fuel, analyses could not be performed for all possible cask/inventory combinations. Since the size of the cask, rather than the details of its construction, determines the size of the cask's inventory, construction details were obtained for one typical spent nuclear fuel transportation cask, and these construction data were the basis for analyses that depended on cask properties. See Appendix B for description and figures of transportation casks.

For base case analyses, the values for the release fractions (f_i) for the five representative elements, cobalt, krypton, cesium, ruthenium, and other (particulate), presented in Table D-21, were taken to be the same as the values presented that were used in the Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel (DOE, 1994d). During the sensitivity studies described below, MACCS calculations were performed that used release fraction values and an inventory for foreign research reactor spent nuclear fuel that were taken from the DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement (Programmatic SNF&INEL Final EIS) (DOE, 1995). Although both the Environmental Assessment and the Programmatic SNF&INEL Final EIS contain release fractions for all six of the severity categories used in the Environmental Assessment, calculations were not performed for the first two categories, because cask failure does not occur for either category, and only a limited number of sensitivity calculations were performed for category 3 because only

Table D-25 Curie Content of Fully Loaded Transportation Casks for Three Representative Fuel Types

Nuclide	Fuel		
	BR-2	RHF	TRIGA
Hydrogen-3	8.6	37	13
Krypton-85	2,470	1,070	363
Strontium-89	40,800	17,600	275
Strontium-Yttrium-90	20,800	8,930	3,160
Yttrium-91	73,000	31,400	4,560
Zirconium-95	107,000	46,300	6,480
Niobium-95	220,000	94,900	12,800
Ruthenium-103, Rh-103m	8,900	3,770	844
Ruthenium-106, Rh-106m	21,500	9,160	2,540
Tin-123	427	184	27
Antimony-125	890	381	119
Tellurium-125m	212	91	29
Tellurium-127m	887	382	56
Tellurium-129m	189	80	23
Cesium-134	16,400	4,000	1,160
Cesium-137	20,600	8,870	3,190
Cerium-141	5,740	2,440	697
Cerium-144	312,000	135,000	25,500
Promethium-147	48,300	24,600	7,020
Promethium-148m	75	29	47
Europium-154	620	163	42
Europium-155	130	46	23
Uranium-234	0.0009	0.0004	0.0001
Uranium-235	0.014	0.01	0.008
Uranium-238	0.0003	0.0002	0.007
Plutonium-238	64	10	3
Plutonium-239	1.8	0.09	0.6
Plutonium-240	1.2	0.4	2
Plutonium-241	284	68	213
Americium-241	0.4	0.1	0.4
Americium-242m	0.001	0.0001	0.009
Americium-243	0.004	0.004	0.0004
Curium-244	1.3	0.009	0.007
Curium-242	1.8	0.1	3

corrosion product are released in a category 3 accident, and only minor amounts of corrosion product deposits form on research reactor spent nuclear fuel. To examine the possible impacts of corrosion products release, during the sensitivity studies, one category 3 accident calculation was performed during which 350 Ci of Co-60 was the only nuclide released, and one calculation was performed that added the same amount of Co-60 to the base case calculation.

D.5.3.1.9 Source Term Timing and Sensible Heat

Ship accident source terms may have both a puff (an immediate release of most material) and a tail (a gradual release of the material over an extended time), where the puff follows the mechanical failure of the cask due to the collision forces, and the tail is produced by the slow heating of the cask contents by an ensuing fire. Because ship collisions are short duration events, if the collision causes a mechanical release, it should be of relatively short duration and the gases released from the cask should be cold (no significant sensible heat content) and thus not subject to plume rise. Conversely, because a substantial engulfing fire that burns for approximately an hour is required to heat both the cask and the spent nuclear fuel elements in the cask to temperatures where cesium compounds (for example, CsOH) vaporize to a significant extent, thermal releases will be delayed (release won't occur until about one hour after the collision) and will not take place rapidly (release duration of about one hour). Of course, if cask failure is caused by thermal rather than mechanical loads, any radioactivity released inside of the cask by the collision will not be released from the cask until the cask fails due to those thermal loads. Moreover, if heat loads cause the fuel elements in the cask to fail at essentially the same time that the cask seals fail due to thermal stress, a delayed short duration release could occur. Thus, ship accident source terms can have four release patterns: (1) a single short (15 minute) release caused by the mechanical forces engendered by the collision; (2) a single short (15 minute) release caused by the mechanical forces engendered by the collision followed by a longer (60 minute) release caused by the thermal loads produced by an ensuing fire; (3) a single long duration (60 minute) release caused by thermal loads on the cask if the collision does not lead to failure but an ensuing fire does; and (4) a single delayed short (15 minute) duration release if cask failure and burst rupture of fuel elements occur together.

Because a substantial engulfing fire of significant duration is required to cause a thermal release, for such thermal releases the radioactivity released from the failed cask will be assumed to be released into the fire plume, which typically will have a bulk gas temperature of about 1,200°K (1,700°F). Therefore, the sensible heat content of that plume will be 100 kilowatts for severity category 5 releases and 150 kilowatts for severity category 6 releases.

The start time and duration of the four release patterns described above are presented in Table D-26. For base case calculations, the first release pattern will be assumed for severity Category 4 accidents and the second pattern for severity Category 5 and 6 accidents. The third and forth release patterns will be examined by sensitivity studies.

Table D-26 Release Timing Patterns

Pattern	*Puff*		*Tail*	
	Release Start (min)	*Release Duration (min)*	*Release Start (min)*	*Release Duration (min)*
1	0	10		
2	0	10	60	60
3			60	60
4			90	10

D.5.3.2 Population Distributions

MACCS calculations require as input a population distribution and site-specific weather conditions. The populations along each of the sixteen compass sectors (N, NNE, NE, etc.) are used to determine the exposed population for each combination of site weather and wind rose conditions. Depending upon the shape of the plume, the exposed population includes the people along one or more adjacent sectors.

The required population distributions were generated for two locations at each of thirteen ports. Table D-27 lists the ports selected for examination in this study.

Table D-27 Ports Analyzed

Coast	High Population	Medium Population	Low Population
East	Philadelphia, PA New York, NY	Hampton Roads, VA Jacksonville, FL	Charleston, SC MOTSU, NC Savannah, GA Wilmington, NC
West	Long Beach, CA	Concord NWS, CA Portland, OR Tacoma, WA	
Gulf			Galveston, TX

Two accident locations were considered for each port, one at dockside and one channel location near the population center where a major ship collision would be possible. Two exceptions were made for ports able to share the same channel accident location due to their close proximity to each other. These exceptions are the Port of Wilmington and MOTSU, NC, as well as the Wando Terminal and the Charleston NWS in greater Charleston, SC. Population distributions were constructed on a compass-sector polar coordinate grid that has eleven radial interval (1.6, 3.2, 4.8, 6.4, 8.0, 16, 32, 48, 64, and 20 km or 1, 2, 3, 4, 5, 10, 20, 30, 40, and 50 mi). The distributions were constructed from 1990 block census data using Sandia's SECPOP90 code (Humphreys et al., 1994). The coordinates of the midpoint of the compass-sector polar coordinate grid were selected by inspection of navigational maps for the ports examined. Table D-28 gives the coordinates of these dockside and channel locations, which represent the selected locations for possible accidents. The population distributions generated by SECPOP90 represent the population in an 80.5 km (50 mi) radius around each potential accident site.

The 26 population distributions constructed (two per port) using SECPOP90 were entered into the site data file for the dockside or channel accident location at each of the thirteen ports. Examination of these files shows that many of the cells in the 26 population distributions are empty because they are covered by water (ocean, rivers, bays, harbor channels).

At many ports, the work force population is probably much larger than the residential population, at least in the commercial area near to the port. Therefore, the work force population was estimated for one port, Elizabeth, and added to the distribution that has been constructed for that port using SECPOP90. Then, during the sensitivity studies, the effect of the work force population on consequences of accidents at Elizabeth was examined.

D.5.3.3 Meteorological Data

MACCS calculations examine all possible combinations of a representative set of weather sequences and a representative set of population distributions. MACCS calculations require a site wind rose, to give the exposure probability of the compass sector population distributions and one year of hourly readings of wind speed, atmospheric stability, and rainfall rate. These data may be recorded either at the accident site or at some not-too-distant meteorological station that has similar meteorology and topography as the accident site. These data are used to determine dispersion as a function of downwind transport distance. Site wind rose and rainfall data were available for each of the ports studied. One year of hourly meteorological data was available from National Weather Service Stations located in the port city for only two of the 13 ports studied. For the other 11 ports, hourly meteorological data recorded at a nearby National Weather Service station was used during the base case calculations. Table D-29 presents the locations of the National Weather Service Stations where the hourly meteorological data files used in this study were recorded, and indicates the ports with which each file was used.

Table D-28 Accident Location Map Coordinates

Port	Location	Description	Coordinates	
			Latitude	Longitude
Elizabeth, NJ (for New York)	Dock	Marine Terminal, Sealand Pier	40°39'35"N	74°08'52"W
	Channel	Narrows	40° 36'29"N	74° 02'21"W
Philadelphia, PA	Dock	Packer Avenue Marine Terminal(container berths)	39° 53'55"N	75° 08'09"W
	Channel	Commodore Berry Fixed Bridge	39° 49'43"N	75° 22'18"W
Norfolk, VA (for Hampton Roads)	Dock	Portsmouth Marine Terminal	36° 51'25"N	76° 19'45"W
	Channel	Willoughby Bank, Northside	36° 59'57"N	76° 18'43"W
MOTSU, NC	Dock	Sunny Point, Wharf 1	33° 59'39"N	77° 51'21"W
	Channel	Lower Swash Channel	33° 54'39"N	78° 01'12"W
Charleston, SC	Dock	Pier at Wando Terminal	32° 49'51"N	79° 53'34"W
	Dock	Naval Weapons Station	32°56'12"N	79°56'11"W
	Channel	Commercial anchorage area D	32° 47'05"N	79° 55'10"W
Savannah, GA	Dock	Savannah Ocean Terminal	32° 05'00"N	81° 05'18"W
	Channel	Intersection Savannah River and Intracoastal Waterway at Elba Island Cut	32° 04'26"N	80° 58'17"W
Galveston, TX	Dock	Container Terminal, Pier 9	29° 19'00"N	94° 46'53"W
	Channel	Cross of Bolivar Roads Channel and Galveston Channel	29° 20'27"N	95° 46'12"W
Concord NWS, CA	Dock	Naval Weapons Station	38° 03'32"N	122° 00'49"W
	Channel	San Francisco Bay Temporary Anchorage No. 7 Shipping Lane Route	37°49'24"N	122°23'47"W
Tacoma, WA	Dock	Port of Tacoma Pier No. 7	47° 16'03"N	122°24'49"W
	Channel	Intersection of 4 shipping lanes in Puget Sound north of Port Townsend	48° 11'24"N	122°49'48"W
Wilmington, NC	Dock	Main Dock Wilmington Terminal	34° 13'03"N	77°57'09"W
	Channel	Lower Swash Channel	33° 54'39"N	78° 01'12"W
Jacksonville, FL	Dock	Blount Island Terminal	30° 23'16"N	81° 33'00"W
	Channel	St. John's River Ferry crossing to Mayport	30° 23'40"N	81° 26'00"W
Long Beach, CA	Dock	Pier E	33° 45'43"N	118° 12'31"W
	Channel	Breakwater East Side	33° 43'23"N	118° 10'53"W
Portland, OR	Dock	Terminal 2	45° 32'54"N	122° 41'56"W
	Channel	St. Johns Bridge	45° 35'04"N	122° 45'58"W

Table D-29 Locations of National Weather Service Stations

Port	National Weather Service Station
Elizabeth, NJ	New York City, NY
Philadelphia, PA	New York City, NY
Norfolk, VA	Cape Hatteras, NC
MOTSU, NC; Wilmington, NC	Cape Hatteras, NC
Charleston, SC; Savannah, GA; Jacksonville, FL	Charleston, SC
Long Beach, CA; Concord NWS, CA	Santa Maria, CA
Portland, OR; Tacoma, WA	Seattle, WA
Galveston, TX	Lake Charles, LA

Although MACCS calculations can use constant meteorology, one year of hourly meteorological data is preferred because adverse results are often the result of meteorological sequences that involve changing meteorological conditions. MACCS uses an importance sampling method to find these less probable sequences that yield adverse results. The sampling method examines all of the 8,760 weather sequences in one year of hourly data and selects the start times of the approximately 100 weather sequences that are used during a variable meteorology calculation. The impact of using constant versus variable meteorology is the subject of one of the sensitivity calculations.

D.5.4 MACCS Calculations

All of the MACCS calculations performed during this study used a source term probability of one. Thus, the consequence estimates generated and the probabilities associated with those estimates are conditional on the release of the source term (i.e., the estimates are conditional on the accident having occurred).

For any source term, a MACCS calculation generates results for all possible combinations of a representative set of weather sequences and a representative set of exposed downwind populations. Since the probability of occurrence of each weather sequence and the exposure probability of each population distribution is known, the variability of consequences due to weather and population conditional on the accident having occurred can be displayed by plotting the probability that a consequence magnitude will be equaled or exceeded against consequence magnitude. Such a plot is called a Complementary Cumulative Distribution Function.

Two types of MACCS accident consequence calculations were performed, base case calculations and sensitivity calculations. Base case calculations used:

- the inventories given in Table D-25,
- the release fractions presented in Table D-21 for severity categories 4, 5, and 6,
- the release timings specified in Table D-26 (pattern 1 was used for severity category 4 releases and pattern 3 for category 5 and 6 releases),
- one year of hourly meteorological data recorded at the National Weather Service Station listed in Table D-29, and
- population distributions calculated using SECPOP90 for the dockside and channel locations presented in Table D-28.

Population distributions and other site-specific data are input to MACCS via a site data file.

Sensitivity calculations modified the input used in the base case calculations to identify the influence on consequences:

- of using variable meteorology recorded offsite at a nearby National Weather Service station rather than constant meteorology recorded onsite at the harbor,
- of using source terms that contained 17 nuclides for which acute health effect dose conversion factors were not available,
- of neglecting the enhanced shielding likely to be afforded to population near to the harbor by the masonry buildings that typify construction in urban commercial neighborhoods,

- of using release fractions developed for the Programmatic SNF&INEL Final EIS (DOE, 1995) for truck and rail accidents,
- of adding the harbor work force population to the residential population distribution,
- of modeling extremely high temperature effects on aluminum-based and TRIGA fuels release fractions,
- of modeling accidents that lead to severe fires using a puff and a tail (two segment release) rather than only a puff, and
- of adding cobalt-60 to the inventory so that corrosion products release can be calculated.

The results of these sensitivity calculations are presented in Section D.5.4.3.

Both the variable meteorology and the constant meteorology MACCS calculations performed for this study consist of a large number of individual trials (about 1,750 trials for each variable meteorology calculation; about 1,150 trials for each constant meteorology calculation). By accumulating the results of the individual trials, an expected (mean) result and a Complementary Cumulative Distribution Function are generated for each output quantity (result) calculated. In addition, for each output calculated, the value of the largest result obtained for any individual trial, the probability of occurrence of that trial, and the weather sequence used in that trial are saved by MACCS.

D.5.4.1 Acute Health Effects

The MACCS code can calculate the numbers of fatalities and injuries that are caused by acute exposures that occur over time periods of a few days (due to dose to the stomach or the intestines) to one year (due to internal dose to the lungs). Of the seven acute injuries that MACCS can examine, prodromal vomiting is the acute injury most likely to appear at low doses and dose rates. Because the occurrence of acute health effects would be cause for considerable concern, acute fatalities and cases of prodromal vomiting were calculated during every MACCS run made during this study; and the output of every run was inspected to see if either acute effect had occurred. Inspection of all of the MACCS output generated showed that no acute fatalities and no cases of prodromal vomiting were ever predicted to occur for any output quantile (i.e., the mean result, all quantile values on the Complementary Cumulative Distribution Function, and the result obtained for the least favorable weather sequence were all zero for acute fatalities and cases of prodromal vomiting).

D.5.4.2 Base Case Calculations

The base case calculations estimated the consequences that might result if any one of nine ship accidents (the combination of three cask inventories presented in Table D-25, with the release fractions for accident severity categories 4, 5, and 6) occurred at any of the 25 accident locations examined (one dockside and one channel location at each of the 13 ports, except MOTSU and Wilmington, which share a channel accident location). Thus, 3 x 3 x 25 = 225 base case MACCS calculations were performed and are presented in this assessment.

D.5.4.2.1 Typical Output

Table D-30 presents MACCS output for one base case calculation, the calculation for the channel accident location at Elizabeth performed with the BR-2 source term and severity categories 4, 5, and 6 release fractions. Using as an example the severity category 5 results, the first group of results in this table are

Table D-30 Sample Output from MACCS

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA4 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5675	4.16E-05	2.43E-07	1.30E-04	2.15E-04	4.38E-04	8.02E-04	1.13E-03	2.50E-04	73
CAN FAT/TOTAL	0-80.5 KM	1.0000	1.64E-04	7.15E-05	4.38E-04	6.29E-04	9.89E-04	1.29E-03	1.50E-03	2.50E-04	73
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.5675	9.45E-04	5.56E-06	2.90E-03	4.96E-03	1.03E-02	1.56E-02	2.56E-02	2.50E-04	73
EDEWBODY TOT LIF	0-8.1 KM	0.8016	2.42E-03	1.01E-03	7.02E-03	9.14E-03	1.60E-02	2.50E-02	3.30E-02	2.50E-04	73
EDEWBODY TOT LIF	0-16.1 KM	0.8848	3.04E-03	1.18E-03	8.37E-03	1.16E-02	2.04E-02	2.59E-02	3.37E-02	2.50E-04	73
EDEWBODY TOT LIF	0-80.5 KM	1.0000	3.76E-03	1.65E-03	9.67E-03	1.33E-02	2.27E-02	3.13E-02	3.41E-02	2.50E-04	73
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	5.98E-07	4.09E-07	1.13E-06	1.48E-06	2.39E-06	NOT-FOUND	3.66E-06	3.45E-03	73
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5675	3.50E-05	2.28E-07	1.11E-04	1.86E-04	3.60E-04	6.54E-04	9.36E-04	2.50E-04	73
CAN FAT/TOTAL	0-80.5 KM	1.0000	1.40E-04	6.32E-05	3.67E-04	5.20E-04	8.26E-04	1.11E-03	1.24E-03	2.50E-04	73
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		1.0000	3.35E-03	1.46E-03	9.03E-03	1.21E-02	2.05E-02	2.58E-02	2.96E-02	2.50E-04	73
TOTAL INGESTION PATHWAYS DOSE		1.0000	5.65E-05	4.28E-05	1.18E-04	1.60E-04	2.50E-04	3.14E-04	3.57E-04	2.24E-05	28
LONG-TERM GROUNDSHINE DOSE		1.0000	2.66E-03	1.15E-03	6.82E-03	9.90E-03	1.43E-02	2.13E-02	2.39E-02	2.50E-04	73
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
CROP DISPOSAL COST		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0000	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA5 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6818	9.85E-02	2.32E-06	2.71E-02	2.60E-01	2.60E+00	6.86E+00	1.75E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	1.0000	2.90E+00	1.22E+00	7.38E+00	1.17E+01	2.91E+01	3.89E+01	5.53E+01	7.65E-05	45
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.6818	2.36E+00	5.22E-05	6.05E-01	6.11E+00	6.21E+01	1.74E+02	4.21E+02	7.65E-05	45
EDEWBODY TOT LIF	0-8.1 KM	0.8854	1.57E+01	2.08E-01	3.35E+01	6.23E+01	3.23E+02	5.13E+02	9.27E+02	9.81E-05	45
EDEWBODY TOT LIF	0-16.1 KM	0.9686	3.30E+01	3.13E+00	9.40E+01	1.48E+02	5.48E+02	8.50E+02	1.25E+03	1.16E-05	44
EDEWBODY TOT LIF	0-80.5 KM	1.0000	6.93E+01	2.84E+01	1.80E+02	2.61E+02	7.09E+02	9.41E+02	1.33E+03	7.65E-05	45
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	1.17E-03	1.60E-06	5.29E-03	6.69E-03	1.52E-02	NOT-FOUND	4.12E-02	1.06E-03	45
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6371	9.83E-02	2.14E-07	2.71E-02	2.60E-01	2.60E+00	6.86E+00	1.75E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	0.9943	2.90E+00	1.22E+00	7.38E+00	1.17E+01	2.88E+01	3.89E+01	5.53E+01	7.65E-05	45
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		0.9943	6.92E+01	2.84E+01	1.80E+02	2.61E+02	7.09E+02	9.41E+02	1.33E+03	7.65E-05	45
TOTAL INGESTION PATHWAYS DOSE		0.9943	1.70E+00	2.50E-01	6.02E+00	8.37E+00	1.07E+01	1.24E+01	1.66E+01	9.94E-06	25
LONG-TERM GROUNDSHINE DOSE		0.9942	6.72E+01	2.61E+01	1.77E+02	2.57E+02	7.09E+02	9.41E+02	1.32E+03	7.65E-05	45
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0038	1.80E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	5.64E+03	2.44E-03	16
CROP DISPOSAL COST		0.0038	1.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	4.15E+03	2.44E-03	16
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0038	6.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	1.61E+00	3.80E-03	16

SITE=NEW LOC=CHANNEL INV=BR-2 ST=EA6 VAR MET=NYC

		PROB NON-ZERO	MEAN	QUANTILES 50TH	90TH	95TH	99TH	99.9TH	PEAK CONS	PEAK PROB	PEAK TRIAL
EARLY + CHRONIC RESULTS											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.6713	8.02E-02	7.79E-07	1.48E-02	1.73E-01	2.59E+00	7.25E+00	1.91E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	1.0000	2.84E+00	1.14E+00	6.92E+00	1.14E+01	3.15E+01	4.36E+01	6.04E+01	7.65E-05	45
POPULATION DOSE (SV)											
EDEWBODY TOT LIF	0-1.6 KM	0.6713	1.92E+00	1.98E-05	3.56E-01	3.03E+00	6.20E+01	1.88E+02	4.59E+02	7.65E-05	45
EDEWBODY TOT LIF	0-8.1 KM	0.8854	1.43E+01	2.64E-02	2.23E+01	4.25E+01	3.31E+02	5.41E+02	1.01E+03	7.65E-05	45
EDEWBODY TOT LIF	0-16.1 KM	0.9686	3.02E+01	5.66E-01	7.75E+01	1.32E+02	5.82E+02	8.62E+02	1.37E+03	1.16E-05	44
EDEWBODY TOT LIF	0-80.5 KM	1.0000	6.77E+01	2.71E+01	1.61E+02	2.57E+02	7.46E+02	1.08E+03	1.45E+03	7.65E-05	45
CENTERLINE DOSE AT SOME DISTANCES (SV)											
EDEWBODY TOT LIF	0-1.6 KM	1.0000	9.53E-04	3.45E-07	1.32E-03	6.25E-03	1.75E-02	NOT-FOUND	4.50E-02	1.06E-03	45
CHRONIC RESULTS ONLY											
HEALTH EFFECTS CASES											
CAN FAT/TOTAL	0-1.6 KM	0.5752	8.01E-02	0.00E+00	1.48E-02	1.73E-01	2.59E+00	7.25E+00	1.91E+01	7.65E-05	45
CAN FAT/TOTAL	0-80.5 KM	0.9879	2.83E+00	1.14E+00	6.92E+00	1.13E+01	3.15E+01	4.36E+01	6.04E+01	7.65E-05	45
EDEWBODY POP. DOSE (SV)	0-80.5 KM										
TOTAL LONG-TERM PATHWAYS DOSE		0.9879	6.76E+01	2.71E+01	1.61E+02	2.57E+02	7.46E+02	1.08E+03	1.45E+03	7.65E-05	45
TOTAL INGESTION PATHWAYS DOSE		0.9879	1.83E+00	2.61E-01	6.23E+00	8.54E+00	1.09E+01	1.29E+01	1.81E+01	9.94E-06	25
LONG-TERM GROUNDSHINE DOSE		0.9879	6.54E+01	2.46E+01	1.59E+02	2.55E+02	7.42E+02	1.06E+03	1.44E+03	7.65E-05	45
ECONOMIC COST MEASURES ($)	0-80.5 KM										
TOTAL ECONOMIC COSTS		0.0038	1.80E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	5.64E+03	2.44E-03	16
CROP DISPOSAL COST		0.0038	1.33E+01	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	4.15E+03	2.44E-03	16
MAXIMUM LONG-TERM ACTION DISTANCE (KM)											
CROP DISPOSAL DIST.		0.0038	6.12E-03	0.00E+00	0.00E+00	0.00E+00	0.00E+00	NOT-FOUND	1.61E+00	3.80E-03	16

Health Effect Cases. The first health effect considered is the number of cancer fatalities expected to occur among the population located within 1.6 km (1 mi) of the accident location. For this population group, the table shows:

- that the probability of getting a nonzero result is 0.6818 which means that not even a fractional cancer fatality was predicted to occur in this population group for 31.82 percent of the approximately 1,750 trials run during this calculation (conversely, at least a fractional cancer death was predicted to occur in 68.18 percent of the trials);
- that the expected (mean) number of cancer fatalities for this population group is 0.098;
- that the 90th and 99th quantiles of the Complementary Cumulative Distribution Function of cancer fatalities for this population group have values of 0.0271 and 2.60; and
- that the largest number of cancer fatalities predicted for this population group for any weather trial was 17.5, that this result had a probability of occurrence of 0.000077, and that the 45th weather sequence selected by the importance sampling scheme led to this result. While the number of LCF is two orders of magnitude higher than the mean, the probability of occurrence of this peak value is four orders of magnitude lower than the mean value.

Figure D-54 presents the Complementary Cumulative Distribution Function for cancer fatalities among the population located within 1.6 km (1 mi) of the channel accident location at Elizabeth, for a severity category 5 accident release fraction. Figure D-54 shows that there is one chance in a thousand (probability = 0.001) that an accident that leads to a severity category 5 release from a cask that contains the BR-2 inventory will produce at least seven cancer deaths. Thus, the 99.9th quantile of the Complementary Cumulative Distribution Function has a value of about seven. Inspection of the Complementary Cumulative Distribution Function also shows that the tail of the distribution has a probability of occurrence of 0.0001 and a magnitude of about 17. These are the values produced by the weather trial that led to the largest result among the full set of weather trials.

The results presented in Table D-30 illustrate a pattern that is general over all of the calculations performed: population dose increases monotonically as distance range increases (e.g., 0-1.6 km, 0-8.1 km, ..., 0-80.5 km). Although not shown in Table D-30, this applies to cancer deaths also. Note that all doses are in Sieverts.

The centerline dose to an individual standing under the plume decreases monotonically with increasing distance, as it should, until it reaches the last computational interval (64.4-80.5 km or 40-50 mi, not shown on D-30) where counter-intuitively it increases. It increases because, during all calculations, rain was artificially forced to occur when the radioactive plume entered this computational interval in order to ensure that all radioactive particulates in the plume deposit onto the ground before the plume exits the computational grid at 80.5 km (50 mi) from the accident location. Deposition of all remaining radioactive particulates onto the ground within the last computational interval ensures that all radioactivity that might enter food pathways at some time after the accident does enter those pathways.

Another pattern that can be seen from Table D-30 is that total population dose is caused almost entirely by long-term groundshine exposures (external direct exposure to radiation emitted by radionuclides deposited on the ground).

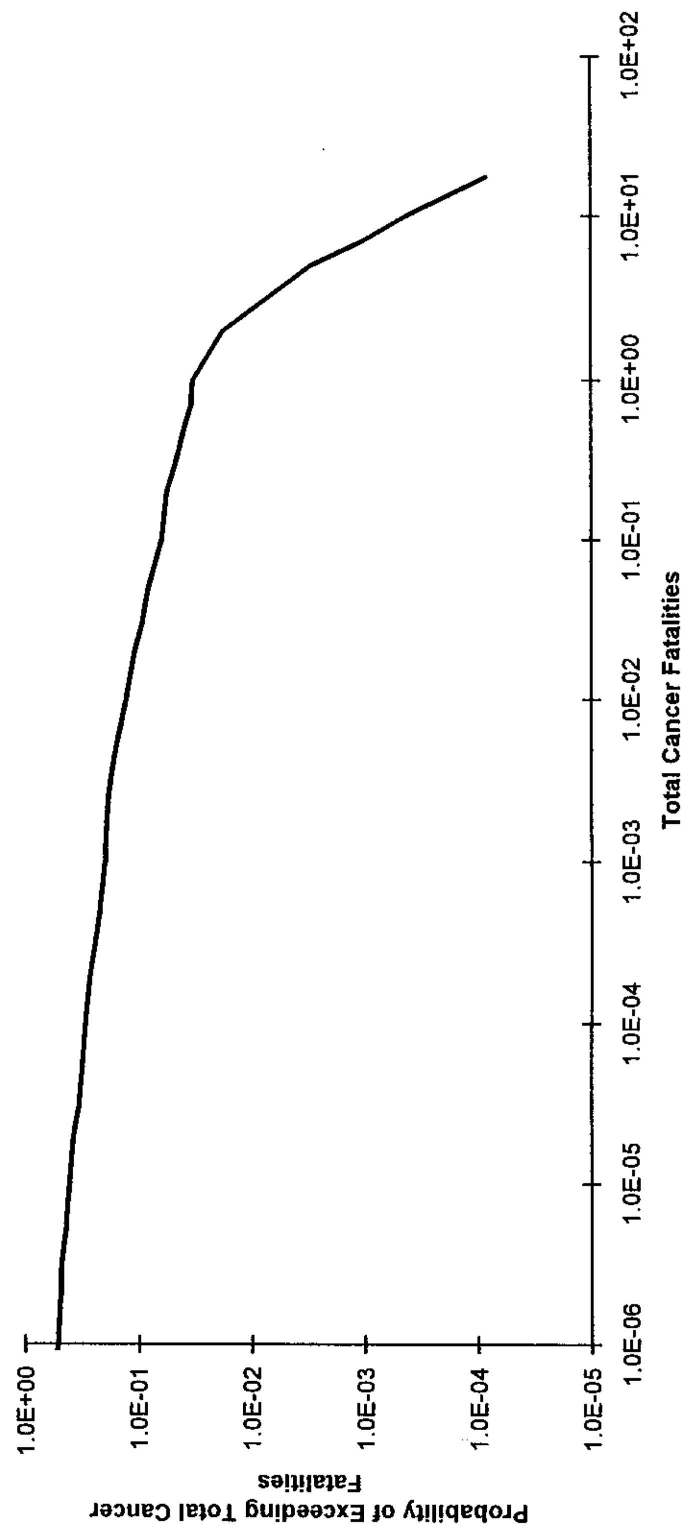


Figure D-54 Total Cancer Fatalities, 0-1.6 km (0-1 mi), Elizabeth Channel, Variable Meteorology, BR-2 Inventory, Severity Category 5

D-30 also shows that the economic losses (costs) caused by the accident are very small (expected value of \$18.00; peak value of \$5,640) and are entirely attributable to the disposal of contaminated crops and milk by farms located close to the accident site (the largest disposal distance found was 1.6 km or 1 mi). This also is typical of the MACCS output for all accidents analyzed.

The values of mean (expected) centerline dose (D_{cl}) (not shown in Table D-30) for severity category 5 release fractions are plotted versus distance (d) in Figure D-55. The figure shows that on a log-log plot dose decreases linearly with distance with a slope very close to minus one. Therefore, as one would expect, individual centerline dose is inversely proportional to distance ($D_{Cl} \propto 1/d$).

Table D-30 presents a breakdown of long-term population dose (calculated as a wholebody dose by the Effective Dose Equivalent method and thus labeled EDEWBODY POP. DOSE) by exposure pathways. Inspection of this breakdown and comparison of the total long-term pathway dose to the total population dose for release category 5, mean results, in the 0-80.5 km (0-50 mi) ranges shows:

- that the total population dose 6,930 rem (69.3 Sv), is almost entirely due to the 6,920 rem (69.2 Sv) dose delivered by long-term exposure pathways;
- that short-term (acute) pathways deliver only a minor dose of 10 rem (0.1 Sv), which is the difference between the 69.3 Sv and the 69.2 Sv;
- that the long-term dose of 6,920 rem (69.2 Sv) is caused mainly by direct exposure pathways [6,750 rem (67.5 Sv)] and only secondarily by ingestion pathways [170 rem (1.7 Sv)];
- that groundshine [6,720 rem (67.2 Sv)] causes almost all of the long-term direct dose; resuspension (external direct exposure to radiation emitted by radionuclides resuspended from the ground) causes the rest of the long-term pathway dose, 30 rem (0.3 Sv);
- that the dose from radioactivity deposited directly on food crops [125 rem (1.25 Sv)] or on grass consumed by milk cows [30 rem (0.30 Sv)] accounts for most ingestion dose; and
- that the rest of the ingestion dose is caused by root uptake [to food crops, 10 rem (0.10 Sv); to grass and fodder crops, 4 rem (0.04 Sv)] with drinking of contaminated water causing only a very small dose of 1 rem (0.01 Sv).

D.5.4.2.2 Principal Base Case Consequence Results

Accident consequence mean (expected) values for whole body population dose and total cancer fatalities for the distance range 0-80.5 km (0-50 mi), and individual centerline dose and individual centerline cancer risk for the distance range 0-1.6 km (0-1 mi) are presented in Table D-31. Table D-32 provides 99.9th quantile values for whole body population dose and total cancer fatalities for the range 0-80.5 km (0-50 mi). Table D-33 presents the largest (peak) result calculated for individual centerline dose and cancer risk in the range 0-1.6 km or 0-1 mi. Table D-34 presents probabilities of the largest results calculated.

Table D-31 shows that the expected total population dose within 80.5 km (50 mi) of the accident location varies from a low of 0.000972 person-rem (0.00000972 person-Sv) for the MOTSU dock calculation that used the TRIGA inventory, severity category 4 (EA4) release fractions, and Cape Hatteras weather to a high of 6,930 person-rem (69.3 person-Sv) for the Elizabeth channel calculation that used the BR-2 inventory, severity category 5 (EA5) release fractions, and New York City weather. Since the total

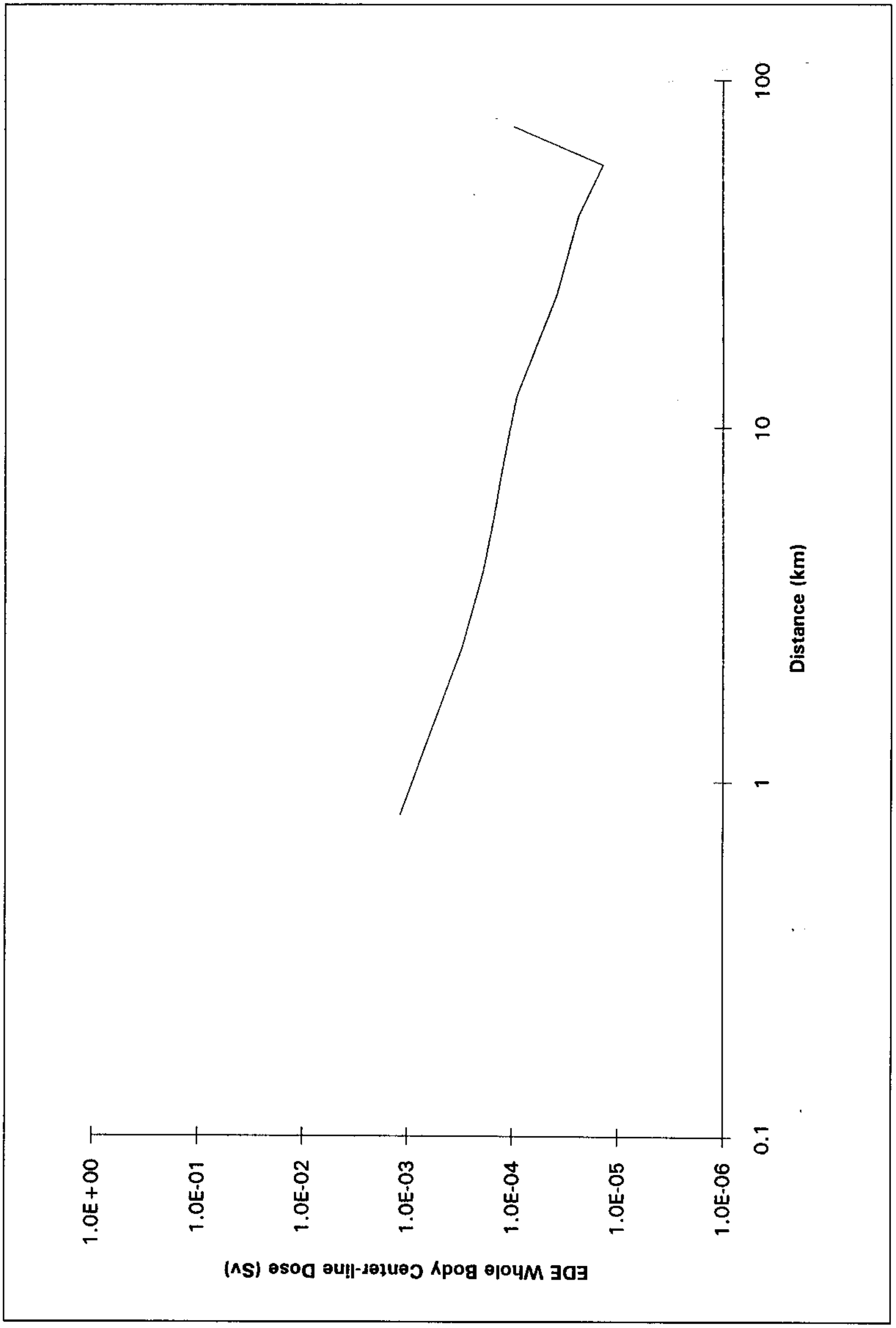


Figure D-55 Mean Effective Dose Equivalent Whole Body Center-Line Dose (Sv) vs Distance, Elizabeth Channel, Variable Meteorology, BR-2 Inventory, Severity Category 5

Table D-31 Mean Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	2.40E-04	4.15E+00	4.13E+00	9.55E-05	1.54E+00	1.53E+00	2.97E-05	5.32E-01	5.26E-01
CHA-C	3.78E-04	4.18E+00	4.21E+00	1.51E-04	1.55E+00	1.56E+00	4.58E-05	5.35E-01	5.37E-01
CNC-D	4.40E-04	2.07E+01	2.21E+01	1.76E-04	7.97E+00	8.51E+00	5.43E-05	2.78E+00	2.97E+00
CNC-C	9.44E-04	3.31E+01	3.40E+01	3.77E-04	1.29E+01	1.32E+01	1.13E-04	4.52E+00	4.63E+00
GAL-D	7.26E-04	1.44E+01	1.58E+01	2.90E-04	5.45E+00	6.00E+00	8.94E-05	1.90E+00	2.08E+00
GAL-C	3.23E-04	1.42E+01	1.55E+01	1.29E-04	5.36E+00	5.89E+00	4.13E-05	1.86E+00	2.04E+00
JAC-D	2.79E-04	6.82E+00	6.76E+00	1.11E-04	2.55E+00	2.52E+00	3.48E-05	8.84E-01	8.71E-01
JAC-C	2.58E-04	5.33E+00	5.45E+00	1.03E-04	1.99E+00	2.03E+00	3.22E-05	6.87E-01	6.99E-01
LOS-D	2.13E-03	4.71E+01	4.82E+01	8.52E-04	1.85E+01	1.89E+01	2.54E-04	6.49E+00	6.62E+00
LOS-C	8.09E-04	4.26E+01	4.40E+01	3.23E-04	1.67E+01	1.73E+01	9.72E-05	5.86E+00	6.05E+00
MOT-D	7.24E-05	2.08E+00	2.21E+00	2.88E-05	7.45E-01	7.91E-01	9.72E-06	2.54E-01	2.70E-01
NEW-D	2.33E-03	6.55E+01	6.51E+01	9.30E-04	2.58E+01	2.56E+01	2.77E-04	9.07E+00	9.00E+00
NEW-C	3.76E-03	6.93E+01	6.77E+01	1.50E-03	2.73E+01	2.67E+01	4.46E-04	9.60E+00	9.37E+00
NOR-D	5.52E-04	8.54E+00	8.32E+00	2.20E-04	3.25E+00	3.15E+00	6.69E-05	1.13E+00	1.09E+00
NOR-C	3.02E-04	6.65E+00	6.64E+00	1.21E-04	2.51E+00	2.50E+00	3.70E-05	8.73E-01	8.67E-01
PHI-D	1.77E-03	2.81E+01	2.78E+01	7.08E-04	1.10E+01	1.08E+01	2.11E-04	3.84E+00	3.79E+00
PHI-C	8.48E-04	2.74E+01	2.81E+01	3.39E-04	1.07E+01	1.09E+01	1.02E-04	3.74E+00	3.83E+00
POR-D	7.70E-04	1.17E+01	1.19E+01	3.07E-04	4.45E+00	4.50E+00	9.32E-05	1.55E+00	1.56E+00
POR-C	5.33E-04	1.12E+01	1.15E+01	2.13E-04	4.26E+00	4.36E+00	6.52E-05	1.48E+00	1.51E+00
SAV-D	5.60E-04	4.91E+00	5.01E+00	2.23E-04	1.80E+00	1.83E+00	6.82E-05	6.18E-01	6.28E-01
SAV-C	1.34E-04	3.82E+00	3.93E+00	5.32E-05	1.38E+00	1.42E+00	1.75E-05	4.74E-01	4.86E-01
SEA-C	1.31E-04	7.54E+00	8.29E+00	5.21E-05	2.84E+00	3.12E+00	1.68E-05	9.86E-01	1.08E+00
TAC-D	5.55E-04	1.73E+01	1.83E+01	2.21E-04	6.67E+00	7.02E+00	6.81E-05	2.33E+00	2.45E+00
TAC-C	3.87E-04	1.43E+01	1.50E+01	1.55E-04	5.50E+00	5.73E+00	4.75E-05	1.92E+00	2.00E+00
WIL-D	3.80E-04	4.82E+00	5.02E+00	1.51E-04	1.79E+00	1.86E+00	4.66E-05	6.19E-01	6.43E-01
WIL-C	9.65E-05	2.07E+00	2.20E+00	3.84E-05	7.47E-01	7.96E-01	1.24E-05	2.56E-01	2.72E-01
CHN-D	1.67E-04	4.76E+00	4.74E+00	6.63E-05	1.76E+00	1.77E+00	2.13E-05	6.09E-01	6.08E-01

Total Cancer Fatalities, 0-80 KM

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	1.05E-05	1.89E-01	1.90E-01	4.20E-06	6.97E-02	6.95E-02	1.24E-06	2.40E-02	2.39E-02
CHA-C	1.66E-05	1.90E-01	1.93E-01	6.65E-06	7.01E-02	7.08E-02	1.90E-06	2.41E-02	2.43E-02
CNC-D	1.91E-05	8.96E-01	9.57E-01	7.63E-06	3.44E-01	3.67E-01	2.23E-06	1.20E-01	1.28E-01
CNC-C	4.10E-05	1.41E+00	1.45E+00	1.65E-05	5.48E-01	5.62E-01	4.63E-06	1.92E-01	1.97E-01
GAL-D	3.17E-05	6.39E-01	7.02E-01	1.27E-05	2.41E-01	2.65E-01	3.70E-06	8.35E-02	9.17E-02
GAL-C	1.39E-05	6.30E-01	6.92E-01	5.57E-06	2.37E-01	2.60E-01	1.71E-06	8.20E-02	9.01E-02
JAC-D	1.22E-05	3.07E-01	3.06E-01	4.88E-06	1.14E-01	1.13E-01	1.45E-06	3.94E-02	3.91E-02
JAC-C	1.13E-05	2.42E-01	2.49E-01	4.51E-06	8.95E-02	9.16E-02	1.34E-06	3.09E-02	3.15E-02
LOS-D	9.32E-05	1.99E+00	2.04E+00	3.75E-05	7.79E-01	7.97E-01	1.04E-05	2.73E-01	2.79E-01
LOS-C	3.51E-05	1.80E+00	1.86E+00	1.41E-05	7.05E-01	7.28E-01	3.96E-06	2.47E-01	2.55E-01
MOT-D	3.16E-06	9.94E-02	1.06E-01	1.25E-06	3.53E-02	3.76E-02	4.13E-07	1.20E-02	1.28E-02
NEW-D	1.02E-04	2.75E+00	2.73E+00	4.09E-05	1.08E+00	1.07E+00	1.13E-05	3.80E-01	3.77E-01
NEW-C	1.64E-04	2.90E+00	2.84E+00	6.62E-05	1.14E+00	1.12E+00	1.83E-05	4.01E-01	3.92E-01
NOR-D	2.42E-05	3.77E-01	3.70E-01	9.71E-06	1.42E-01	1.39E-01	2.76E-06	4.94E-02	4.82E-02
NOR-C	1.32E-05	2.96E-01	2.97E-01	5.30E-06	1.11E-01	1.11E-01	1.53E-06	3.85E-02	3.84E-02
PHI-D	7.75E-05	1.20E+00	1.19E+00	3.12E-05	4.66E-01	4.61E-01	8.67E-06	1.63E-01	1.61E-01
PHI-C	3.70E-05	1.17E+00	1.20E+00	1.49E-05	4.53E-01	4.66E-01	4.19E-06	1.59E-01	1.63E-01
POR-D	3.37E-05	5.18E-01	5.27E-01	1.35E-05	1.95E-01	1.98E-01	3.85E-06	6.78E-02	6.87E-02
POR-C	2.33E-05	4.97E-01	5.12E-01	9.34E-06	1.87E-01	1.92E-01	2.70E-06	6.50E-02	6.66E-02
SAV-D	2.46E-05	2.28E-01	2.34E-01	9.87E-06	8.28E-02	8.46E-02	2.83E-06	2.84E-02	2.90E-02
SAV-C	5.88E-06	1.79E-01	1.85E-01	2.32E-06	6.45E-02	6.65E-02	7.42E-07	2.20E-02	2.27E-02
SEA-C	5.61E-06	3.37E-01	3.70E-01	2.23E-06	1.26E-01	1.39E-01	6.93E-07	4.36E-02	4.79E-02
TAC-D	2.41E-05	7.54E-01	7.95E-01	9.66E-06	2.88E-01	3.04E-01	2.81E-06	1.00E-01	1.06E-01
TAC-C	1.68E-05	6.26E-01	6.55E-01	6.75E-06	2.38E-01	2.49E-01	1.95E-06	8.30E-02	8.66E-02
WIL-D	1.66E-05	2.19E-01	2.29E-01	6.66E-06	8.09E-02	8.43E-02	1.93E-06	2.79E-02	2.90E-02
WIL-C	4.22E-06	9.76E-02	1.04E-01	1.68E-06	3.50E-02	3.74E-02	5.22E-07	1.20E-02	1.28E-02
CHN-D	6.76E-06	2.17E-01	2.19E-01	2.67E-06	7.98E-02	8.03E-02	8.44E-07	2.75E-02	2.76E-02

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-31 Mean Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
CHA-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
CNC-D	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
CNC-C	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
GAL-D	9.29E-07	6.52E-04	6.91E-04	3.71E-07	2.58E-04	2.74E-04	1.05E-07	9.08E-05	9.62E-05
GAL-C	9.29E-07	6.52E-04	6.91E-04	3.71E-07	2.58E-04	2.74E-04	1.05E-07	9.08E-05	9.62E-05
JAC-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
JAC-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
LOS-D	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
LOS-C	1.07E-06	2.28E-04	2.17E-04	4.29E-07	9.01E-05	8.59E-05	1.21E-07	3.17E-05	3.02E-05
MOT-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
NEW-D	5.98E-07	1.17E-03	9.53E-04	2.39E-07	4.63E-04	3.77E-04	6.81E-08	1.63E-04	1.33E-04
NEW-C	5.98E-07	1.17E-03	9.53E-04	2.39E-07	4.63E-04	3.77E-04	6.81E-08	1.63E-04	1.33E-04
NOR-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
NOR-C	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
PHI-D	1.01E-06	6.31E-04	6.59E-04	4.02E-07	2.50E-04	2.61E-04	1.14E-07	8.78E-05	9.16E-05
PHI-C	1.01E-06	6.31E-04	6.59E-04	4.02E-07	2.50E-04	2.61E-04	1.14E-07	8.78E-05	9.16E-05
POR-D	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
POR-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
SAV-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
SAV-C	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05
SEA-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
TAC-D	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
TAC-C	7.54E-07	7.56E-04	7.97E-04	3.01E-07	2.99E-04	3.15E-04	8.57E-08	1.05E-04	1.11E-04
WIL-D	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
WIL-C	5.32E-07	6.24E-04	5.51E-04	2.12E-07	2.47E-04	2.18E-04	6.04E-08	8.69E-05	7.67E-05
CHN-D	8.60E-07	6.83E-04	7.10E-04	3.44E-07	2.70E-04	2.81E-04	9.71E-08	9.51E-05	9.88E-05

Individual Center-line Cancer Risk, 0-1.6 KM

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
CHA-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
CNC-D	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
CNC-C	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
GAL-D	4.43E-08	2.72E-05	2.88E-05	1.80E-08	1.08E-05	1.14E-05	4.63E-09	3.78E-06	4.01E-06
GAL-C	4.43E-08	2.72E-05	2.88E-05	1.80E-08	1.08E-05	1.14E-05	4.63E-09	3.78E-06	4.01E-06
JAC-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
JAC-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
LOS-D	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
LOS-C	5.12E-08	9.50E-06	9.06E-06	2.08E-08	3.75E-06	3.58E-06	5.36E-09	1.32E-06	1.26E-06
MOT-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
NEW-D	2.80E-08	4.88E-05	3.97E-05	1.14E-08	1.93E-05	1.57E-05	2.96E-09	6.78E-06	5.52E-06
NEW-C	2.80E-08	4.88E-05	3.97E-05	1.14E-08	1.93E-05	1.57E-05	2.96E-09	6.78E-06	5.52E-06
NOR-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
NOR-C	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
PHI-D	4.80E-08	2.63E-05	2.75E-05	1.95E-08	1.04E-05	1.09E-05	5.01E-09	3.66E-06	3.82E-06
PHI-C	4.80E-08	2.63E-05	2.75E-05	1.95E-08	1.04E-05	1.09E-05	5.01E-09	3.66E-06	3.82E-06
POR-D	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
POR-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
SAV-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
SAV-C	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06
SEA-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
TAC-D	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
TAC-C	3.55E-08	3.15E-05	3.32E-05	1.44E-08	1.25E-05	1.31E-05	3.74E-09	4.38E-06	4.62E-06
WIL-D	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
WIL-C	2.51E-08	2.60E-05	2.30E-05	1.02E-08	1.03E-05	9.08E-06	2.64E-09	3.62E-06	3.19E-06
CHN-D	4.10E-08	2.85E-05	2.96E-05	1.66E-08	1.13E-05	1.17E-05	4.29E-09	3.96E-06	4.11E-06

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-32 99.9th Quantile Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	1.20E-03	4.63E+01	5.26E+01	3.82E-04	1.71E+01	2.04E+01	1.30E-04	6.21E+00	7.13E+00
CHA-C	3.40E-03	9.03E+01	9.75E+01	1.23E-03	3.70E+01	4.13E+01	3.98E-04	1.22E+01	1.37E+01
CNC-D	3.31E-03	9.47E+01	1.02E+02	1.29E-03	3.41E+01	3.56E+01	3.87E-04	1.17E+01	1.22E+01
CNC-C	1.06E-02	1.55E+02	1.73E+02	3.77E-03	5.55E+01	5.20E+01	1.17E-03	1.91E+01	1.97E+01
GAL-D	5.03E-03	1.33E+02	1.38E+02	2.01E-03	4.90E+01	4.92E+01	5.35E-04	1.66E+01	1.92E+01
GAL-C	1.37E-03	8.70E+01	1.16E+02	6.77E-04	3.72E+01	3.93E+01	2.01E-04	1.30E+01	1.38E+01
JAC-D	1.29E-03	7.23E+01	7.86E+01	5.59E-04	2.78E+01	3.07E+01	1.45E-04	1.01E+01	1.05E+01
JAC-C	1.26E-03	6.40E+01	6.77E+01	5.28E-04	2.45E+01	2.69E+01	1.55E-04	8.55E+00	9.22E+00
LOS-D	NOT-FOUND	2.67E+02	2.70E+02	NOT-FOUND	1.02E+02	1.04E+02	NOT-FOUND	3.32E+01	3.64E+01
LOS-C	3.66E-03	2.19E+02	2.41E+02	1.38E-03	9.72E+01	9.84E+01	5.13E-04	3.14E+01	3.12E+01
MOT-D	5.03E-04	2.46E+01	2.63E+01	1.80E-04	8.87E+00	9.48E+00	6.25E-05	2.90E+00	2.95E+00
NEW-D	1.20E-02	5.87E+02	6.37E+02	4.24E-03	2.42E+02	2.52E+02	1.34E-03	8.29E+01	8.68E+01
NEW-C	3.13E-02	9.41E+02	1.08E+03	1.19E-02	3.86E+02	3.99E+02	3.52E-03	1.33E+02	1.42E+02
NOR-D	3.40E-03	1.03E+02	1.10E+02	1.40E-03	4.04E+01	4.41E+01	4.59E-04	1.30E+01	1.39E+01
NOR-C	1.62E-03	9.02E+01	1.00E+02	7.13E-04	3.35E+01	3.56E+01	2.06E-04	1.16E+01	1.24E+01
PHI-D	8.45E-03	3.10E+02	3.32E+02	3.45E-03	1.15E+02	1.18E+02	1.00E-03	4.65E+01	5.03E+01
PHI-C	5.07E-03	2.86E+02	2.98E+02	2.03E-03	1.05E+02	1.08E+02	6.95E-04	4.13E+01	4.63E+01
POR-D	3.16E-03	1.09E+02	1.13E+02	1.18E-03	4.62E+01	4.90E+01	3.55E-04	1.40E+01	1.59E+01
POR-C	3.48E-03	1.01E+02	1.04E+02	1.35E-03	3.55E+01	4.37E+01	4.02E-04	1.19E+01	1.46E+01
SAV-D	5.26E-03	1.26E+02	1.28E+02	2.10E-03	5.56E+01	5.98E+01	6.05E-04	2.04E+01	2.16E+01
SAV-C	8.01E-04	4.37E+01	4.44E+01	3.33E-04	1.59E+01	1.72E+01	9.87E-05	5.65E+00	5.42E+00
SEA-C	8.69E-04	5.53E+01	5.99E+01	3.54E-04	2.10E+01	2.29E+01	1.12E-04	7.43E+00	8.11E+00
TAC-D	1.55E-03	8.36E+01	8.67E+01	7.01E-04	3.51E+01	3.60E+01	2.09E-04	1.19E+01	1.23E+01
TAC-C	1.87E-03	9.69E+01	1.01E+02	7.23E-04	3.65E+01	4.21E+01	2.20E-04	1.29E+01	1.52E+01
WIL-D	4.47E-03	7.09E+01	7.65E+01	1.79E-03	2.75E+01	2.98E+01	5.10E-04	9.33E+00	1.08E+01
WIL-C	1.03E-03	2.25E+01	2.79E+01	3.89E-04	8.70E+00	9.96E+00	1.18E-04	3.00E+00	3.11E+00
CHN-D	7.70E-04	3.69E+01	3.91E+01	3.04E-04	1.36E+01	1.42E+01	9.88E-05	5.22E+00	5.46E+00

Total Cancer Fatalities, 0-80 KM

Site/Loc	BR-2 EA4	BR-2 EA5	BR-2 EA6	RHF EA4	RHF EA5	RHF EA6	TRIGA EA4	TRIGA EA5	TRIGA EA6
CHA-D	4.33E-05	1.98E+00	2.24E+00	1.63E-05	7.81E-01	8.41E-01	5.50E-06	2.67E-01	2.89E-01
CHA-C	1.30E-04	3.96E+00	4.27E+00	7.14E-05	1.61E+00	1.69E+00	1.60E-05	4.94E-01	5.80E-01
CNC-D	1.43E-04	4.07E+00	4.38E+00	5.84E-05	1.28E+00	1.35E+00	1.60E-05	5.30E-01	5.55E-01
CNC-C	4.60E-04	6.20E+00	6.23E+00	1.80E-04	2.55E+00	2.59E+00	5.25E-05	8.80E-01	8.78E-01
GAL-D	2.07E-04	4.93E+00	5.12E+00	7.64E-05	1.96E+00	2.07E+00	2.18E-05	7.00E-01	7.31E-01
GAL-C	7.04E-05	3.88E+00	4.09E+00	2.89E-05	1.49E+00	1.54E+00	7.95E-06	5.73E-01	5.88E-01
JAC-D	5.80E-05	3.10E+00	3.21E+00	2.38E-05	1.12E+00	1.15E+00	6.67E-06	3.91E-01	4.34E-01
JAC-C	6.04E-05	2.71E+00	2.87E+00	2.29E-05	1.08E+00	1.14E+00	6.50E-06	3.53E-01	3.75E-01
LOS-D	NOT-FOUND	1.03E+01	1.05E+01	NOT-FOUND	4.49E+00	4.52E+00	NOT-FOUND	1.45E+00	1.58E+00
LOS-C	1.98E-04	9.81E+00	1.01E+01	7.39E-05	3.39E+00	3.90E+00	2.08E-05	1.18E+00	1.19E+00
MOT-D	2.32E-05	1.22E+00	1.33E+00	8.61E-06	4.08E-01	4.42E-01	2.67E-06	1.43E-01	1.47E-01
NEW-D	4.40E-04	2.46E+01	2.58E+01	1.72E-04	1.10E+01	1.15E+01	5.49E-05	3.48E+00	3.83E+00
NEW-C	1.29E-03	3.89E+01	4.36E+01	5.34E-04	1.47E+01	1.67E+01	1.40E-04	5.75E+00	6.28E+00
NOR-D	1.55E-04	4.23E+00	4.63E+00	6.51E-05	1.62E+00	1.81E+00	1.85E-05	5.99E-01	6.50E-01
NOR-C	7.23E-05	3.59E+00	3.97E+00	3.08E-05	1.39E+00	1.43E+00	8.46E-06	5.25E-01	5.47E-01
PHI-D	3.54E-04	1.18E+01	1.23E+01	1.29E-04	5.18E+00	5.55E+00	3.88E-05	1.90E+00	2.05E+00
PHI-C	2.03E-04	1.22E+01	1.12E+01	9.95E-05	4.74E+00	4.93E+00	2.85E-05	1.78E+00	1.85E+00
POR-D	1.47E-04	4.90E+00	5.16E+00	6.01E-05	1.96E+00	2.03E+00	1.61E-05	6.89E-01	7.13E-01
POR-C	1.48E-04	3.78E+00	4.71E+00	6.08E-05	1.56E+00	1.84E+00	1.63E-05	5.45E-01	6.19E-01
SAV-D	2.23E-04	5.68E+00	6.22E+00	1.01E-04	2.20E+00	2.45E+00	2.41E-05	7.91E-01	8.82E-01
SAV-C	3.52E-05	1.79E+00	1.93E+00	1.28E-05	6.84E-01	7.04E-01	3.71E-06	2.45E-01	2.62E-01
SEA-C	3.73E-05	2.41E+00	2.61E+00	1.40E-05	9.06E-01	9.68E-01	4.16E-06	3.17E-01	3.47E-01
TAC-D	7.55E-05	3.60E+00	3.74E+00	3.03E-05	1.32E+00	1.42E+00	8.41E-06	5.55E-01	5.74E-01
TAC-C	8.26E-05	3.94E+00	4.73E+00	3.19E-05	1.59E+00	1.87E+00	9.47E-06	5.25E-01	6.20E-01
WIL-D	1.99E-04	2.90E+00	3.25E+00	7.68E-05	1.17E+00	1.21E+00	2.05E-05	3.98E-01	4.27E-01
WIL-C	4.89E-05	1.04E+00	1.16E+00	1.96E-05	4.13E-01	4.66E-01	5.15E-06	1.45E-01	1.55E-01
CHN-D	3.33E-05	1.55E+00	1.70E+00	1.32E-05	6.28E-01	7.15E-01	3.67E-06	2.19E-01	2.32E-01

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-33 Peak Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.82E-03	1.04E+02	1.10E+02	1.12E-03	4.07E+01	4.32E+01	3.41E-04	1.43E+01	1.51E+01
CHA-C	4.41E-03	2.12E+02	2.31E+02	1.76E-03	8.56E+01	9.34E+01	5.23E-04	3.00E+01	3.27E+01
CNC-D	3.35E-03	1.02E+02	1.11E+02	1.33E-03	4.03E+01	4.40E+01	4.00E-04	1.42E+01	1.55E+01
CNC-C	1.42E-02	6.71E+02	7.27E+02	5.70E-03	2.66E+02	2.88E+02	1.72E-03	9.34E+01	1.01E+02
GAL-D	5.60E-03	2.33E+02	2.54E+02	2.22E-03	9.59E+01	1.05E+02	6.69E-04	3.36E+01	3.66E+01
GAL-C	2.09E-03	1.85E+02	2.02E+02	8.28E-04	7.23E+01	7.89E+01	2.56E-04	2.53E+01	2.76E+01
JAC-D	2.15E-03	1.10E+02	1.20E+02	8.53E-04	4.28E+01	4.66E+01	2.61E-04	1.50E+01	1.63E+01
JAC-C	1.90E-03	1.12E+02	1.23E+02	7.52E-04	4.59E+01	5.01E+01	2.31E-04	1.61E+01	1.76E+01
LOS-D	1.30E-02	3.94E+02	4.29E+02	5.20E-03	1.56E+02	1.70E+02	1.54E-03	5.49E+01	5.98E+01
LOS-C	5.97E-03	3.47E+02	3.76E+02	2.39E-03	1.37E+02	1.49E+02	7.21E-04	4.83E+01	5.23E+01
MOT-D	7.72E-04	4.61E+01	5.03E+01	2.98E-04	1.71E+01	1.86E+01	9.74E-05	5.88E+00	6.42E+00
NEW-D	1.86E-02	1.13E+03	1.24E+03	7.44E-03	4.48E+02	4.89E+02	2.24E-03	1.58E+02	1.72E+02
NEW-C	3.41E-02	1.33E+03	1.45E+03	1.36E-02	5.25E+02	5.73E+02	4.01E-03	1.85E+02	2.02E+02
NOR-D	4.29E-03	1.66E+02	1.81E+02	1.71E-03	6.49E+01	7.06E+01	5.08E-04	2.27E+01	2.47E+01
NOR-C	2.78E-03	1.64E+02	1.78E+02	1.11E-03	6.44E+01	7.00E+01	3.36E-04	2.26E+01	2.45E+01
PHI-D	1.13E-02	7.15E+02	7.80E+02	4.52E-03	2.82E+02	3.08E+02	1.36E-03	9.91E+01	1.08E+02
PHI-C	8.77E-03	4.87E+02	5.31E+02	3.50E-03	1.92E+02	2.10E+02	1.06E-03	6.75E+01	7.36E+01
POR-D	3.85E-03	1.69E+02	1.85E+02	1.54E-03	6.69E+01	7.30E+01	4.64E-04	2.35E+01	2.57E+01
POR-C	3.94E-03	1.70E+02	1.85E+02	1.57E-03	6.66E+01	7.27E+01	4.67E-04	2.34E+01	2.55E+01
SAV-D	6.18E-03	2.39E+02	2.60E+02	2.46E-03	9.64E+01	1.05E+02	7.32E-04	3.38E+01	3.69E+01
SAV-C	1.43E-03	7.65E+01	8.00E+01	5.66E-04	2.94E+01	3.07E+01	1.75E-04	1.03E+01	1.07E+01
SEA-C	9.93E-04	5.62E+01	6.13E+01	3.93E-04	2.14E+01	2.33E+01	1.22E-04	7.44E+00	8.11E+00
TAC-D	2.67E-03	1.18E+02	1.29E+02	1.06E-03	4.66E+01	5.09E+01	3.23E-04	1.64E+01	1.79E+01
TAC-C	3.13E-03	1.70E+02	1.85E+02	1.25E-03	6.64E+01	7.25E+01	3.78E-04	2.33E+01	2.54E+01
WIL-D	5.50E-03	1.49E+02	1.63E+02	2.20E-03	5.91E+01	6.44E+01	6.51E-04	2.08E+01	2.27E+01
WIL-C	1.18E-03	3.28E+01	3.58E+01	4.71E-04	1.29E+01	1.41E+01	1.41E-04	4.57E+00	4.99E+00
CHN-D	1.76E-03	6.83E+01	7.43E+01	6.97E-04	2.64E+01	2.87E+01	2.14E-04	9.20E+00	1.00E+01

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.20E-04	4.41E+00	4.69E+00	4.80E-05	1.72E+00	1.83E+00	1.37E-05	6.02E-01	6.39E-01
CHA-C	1.95E-04	8.90E+00	9.71E+00	7.83E-05	3.60E+00	3.93E+00	2.17E-05	1.26E+00	1.38E+00
CNC-D	1.49E-04	4.31E+00	4.68E+00	5.96E-05	1.68E+00	1.83E+00	1.68E-05	5.90E-01	6.44E-01
CNC-C	6.01E-04	2.80E+01	3.03E+01	2.42E-04	1.11E+01	1.20E+01	6.84E-05	3.89E+00	4.22E+00
GAL-D	2.48E-04	9.97E+00	1.09E+01	9.90E-05	4.08E+00	4.45E+00	2.79E-05	1.43E+00	1.55E+00
GAL-C	9.02E-05	7.89E+00	8.61E+00	3.57E-05	3.07E+00	3.35E+00	1.05E-05	1.07E+00	1.17E+00
JAC-D	9.25E-05	4.71E+00	5.12E+00	3.68E-05	1.82E+00	1.98E+00	1.07E-05	6.37E-01	6.93E-01
JAC-C	8.20E-05	4.71E+00	5.13E+00	3.26E-05	1.93E+00	2.11E+00	9.47E-06	6.77E-01	7.38E-01
LOS-D	5.75E-04	1.64E+01	1.79E+01	2.31E-04	6.50E+00	7.08E+00	6.37E-05	2.29E+00	2.49E+00
LOS-C	2.52E-04	1.45E+01	1.57E+01	1.01E-04	5.73E+00	6.20E+00	2.87E-05	2.01E+00	2.18E+00
MOT-D	3.49E-05	2.10E+00	2.29E+00	1.34E-05	7.69E-01	8.39E-01	4.24E-06	2.65E-01	2.88E-01
NEW-D	7.88E-04	4.72E+01	5.16E+01	3.17E-04	1.87E+01	2.04E+01	8.94E-05	6.56E+00	7.16E+00
NEW-C	1.50E-03	5.53E+01	6.04E+01	6.06E-04	2.19E+01	2.39E+01	1.66E-04	7.69E+00	8.39E+00
NOR-D	1.89E-04	7.09E+00	7.71E+00	7.62E-05	2.75E+00	2.99E+00	2.10E-05	9.63E-01	1.05E+00
NOR-C	1.19E-04	6.92E+00	7.52E+00	4.75E-05	2.71E+00	2.94E+00	1.35E-05	9.50E-01	1.03E+00
PHI-D	4.79E-04	2.99E+01	3.26E+01	1.93E-04	1.18E+01	1.29E+01	5.44E-05	4.14E+00	4.52E+00
PHI-C	3.71E-04	2.04E+01	2.23E+01	1.49E-04	8.03E+00	8.77E+00	4.23E-05	2.82E+00	3.08E+00
POR-D	1.68E-04	7.05E+00	7.70E+00	6.74E-05	2.79E+00	3.04E+00	1.87E-05	9.80E-01	1.07E+00
POR-C	1.74E-04	7.18E+00	7.84E+00	7.00E-05	2.81E+00	3.06E+00	1.94E-05	9.84E-01	1.07E+00
SAV-D	2.74E-04	1.00E+01	1.09E+01	1.10E-04	4.07E+00	4.44E+00	3.04E-05	1.43E+00	1.55E+00
SAV-C	6.15E-05	3.32E+00	3.48E+00	2.44E-05	1.27E+00	1.33E+00	7.14E-06	4.42E-01	4.62E-01
SEA-C	4.27E-05	2.47E+00	2.69E+00	1.69E-05	9.33E-01	1.02E+00	4.99E-06	3.24E-01	3.53E-01
TAC-D	1.14E-04	5.03E+00	5.49E+00	4.54E-05	1.95E+00	2.12E+00	1.31E-05	6.83E-01	7.45E-01
TAC-C	1.33E-04	7.18E+00	7.84E+00	5.34E-05	2.80E+00	3.06E+00	1.51E-05	9.82E-01	1.07E+00
WIL-D	2.43E-04	6.24E+00	6.80E+00	9.80E-05	2.46E+00	2.69E+00	2.69E-05	8.70E-01	9.48E-01
WIL-C	5.22E-05	1.41E+00	1.54E+00	2.10E-05	5.42E-01	5.92E-01	5.94E-06	1.93E-01	2.10E-01
CHN-D	7.30E-05	2.95E+00	3.21E+00	2.90E-05	1.13E+00	1.23E+00	8.51E-06	3.95E-01	4.29E-01

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-33 Peak Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
CHA-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
CNC-D	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
CNC-C	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
GAL-D	3.66E-06	5.34E-02	5.82E-02	1.46E-06	2.11E-02	2.31E-02	4.06E-07	7.43E-03	8.10E-03
GAL-C	3.66E-06	5.34E-02	5.82E-02	1.46E-06	2.11E-02	2.31E-02	4.06E-07	7.43E-03	8.10E-03
JAC-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
JAC-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
LOS-D	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
LOS-C	3.66E-06	2.51E-02	2.74E-02	1.46E-06	9.94E-03	1.08E-02	4.06E-07	3.50E-03	3.81E-03
MOT-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
NEW-D	3.66E-06	4.12E-02	4.50E-02	1.46E-06	1.63E-02	1.78E-02	4.06E-07	5.74E-03	6.26E-03
NEW-C	3.66E-06	4.12E-02	4.50E-02	1.46E-06	1.63E-02	1.78E-02	4.06E-07	5.74E-03	6.26E-03
NOR-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
NOR-C	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
PHI-D	3.66E-06	6.06E-02	6.61E-02	1.46E-06	2.40E-02	2.62E-02	4.06E-07	8.43E-03	9.20E-03
PHI-C	3.66E-06	6.06E-02	6.61E-02	1.46E-06	2.40E-02	2.62E-02	4.06E-07	8.43E-03	9.20E-03
POR-D	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
POR-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
SAV-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
SAV-C	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03
SEA-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
TAC-D	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
TAC-C	3.66E-06	5.64E-02	6.15E-02	1.46E-06	2.23E-02	2.43E-02	4.06E-07	7.85E-03	8.56E-03
WIL-D	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
WIL-C	3.66E-06	5.96E-02	6.50E-02	1.46E-06	2.36E-02	2.57E-02	4.06E-07	8.29E-03	9.04E-03
CHN-D	3.66E-06	5.85E-02	6.39E-02	1.46E-06	2.32E-02	2.53E-02	4.06E-07	8.15E-03	8.89E-03

Individual Center-line Cancer Risk, 0-1.6 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
CHA-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
CNC-D	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
CNC-C	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
GAL-D	1.79E-07	2.23E-03	2.43E-03	7.29E-08	8.80E-04	9.60E-04	1.84E-08	3.09E-04	3.37E-04
GAL-C	1.79E-07	2.23E-03	2.43E-03	7.29E-08	8.80E-04	9.60E-04	1.84E-08	3.09E-04	3.37E-04
JAC-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
JAC-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
LOS-D	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
LOS-C	1.79E-07	1.05E-03	1.14E-03	7.29E-08	4.14E-04	4.52E-04	1.84E-08	1.46E-04	1.59E-04
MOT-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
NEW-D	1.79E-07	1.72E-03	1.88E-03	7.29E-08	6.80E-04	7.42E-04	1.84E-08	2.39E-04	2.61E-04
NEW-C	1.79E-07	1.72E-03	1.88E-03	7.29E-08	6.80E-04	7.42E-04	1.84E-08	2.39E-04	2.61E-04
NOR-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
NOR-C	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
PHI-D	1.79E-07	2.53E-03	2.76E-03	7.29E-08	9.98E-04	1.09E-03	1.84E-08	3.51E-04	3.83E-04
PHI-C	1.79E-07	2.53E-03	2.76E-03	7.29E-08	9.98E-04	1.09E-03	1.84E-08	3.51E-04	3.83E-04
POR-D	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
POR-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
SAV-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
SAV-C	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04
SEA-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
TAC-D	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
TAC-C	1.79E-07	2.35E-03	2.57E-03	7.29E-08	9.29E-04	1.01E-03	1.84E-08	3.27E-04	3.56E-04
WIL-D	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
WIL-C	1.79E-07	2.48E-03	2.71E-03	7.29E-08	9.82E-04	1.07E-03	1.84E-08	3.45E-04	3.77E-04
CHN-D	1.79E-07	2.44E-03	2.66E-03	7.29E-08	9.65E-04	1.05E-03	1.84E-08	3.39E-04	3.70E-04

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-34 Probability of Peak Results, Variable Meteorology

EDE Whole Body Population Dose, 0-80 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
CHA-C	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06
CNC-D	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05
CNC-C	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06
GAL-D	7.80E-06	9.75E-06	9.75E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06
GAL-C	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06
JAC-D	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06
JAC-C	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06
LOS-D	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06
LOS-C	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06
MOT-D	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05
NEW-D	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05
NEW-C	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05
NOR-D	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06
NOR-C	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05
PHI-D	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06
PHI-C	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05
POR-D	1.26E-05	1.09E-05	1.09E-05	1.26E-05	1.09E-05	1.09E-05	1.26E-05	1.09E-05	1.09E-05
POR-C	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05
SAV-D	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06
SAV-C	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05
SEA-C	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04	1.36E-04	9.80E-04	9.80E-04
TAC-D	5.33E-06	1.14E-05	1.14E-05	5.33E-06	1.14E-05	1.14E-05	5.33E-06	1.14E-05	1.14E-05
TAC-C	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05
WIL-D	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04
WIL-C	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04
CHN-D	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06

Total Cancer Fatalities, 0-80 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
CHA-C	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06	7.77E-06	4.44E-06	4.44E-06
CNC-D	8.47E-04	5.08E-04	5.08E-04	8.47E-04	1.18E-05	1.18E-05	8.47E-04	1.18E-05	1.18E-05
CNC-C	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06	7.58E-06
GAL-D	7.80E-06	9.75E-06	9.75E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06	7.80E-06
GAL-C	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06	7.80E-06	9.75E-06	9.75E-06
JAC-D	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06	6.03E-06
JAC-C	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06	6.08E-06	8.06E-06	8.06E-06
LOS-D	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06	1.83E-03	3.42E-06	3.42E-06
LOS-C	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06	4.91E-06	2.81E-06	2.81E-06
MOT-D	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05	1.67E-05	2.76E-05	2.76E-05
NEW-D	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05	2.03E-05
NEW-C	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05	2.50E-04	7.65E-05	7.65E-05
NOR-D	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06	3.51E-04	9.62E-06	9.62E-06
NOR-C	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05	1.07E-05	1.61E-05	1.61E-05
PHI-D	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06	3.17E-06
PHI-C	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05	2.44E-05
POR-D	2.98E-04	1.09E-05	1.09E-05	2.98E-04	1.09E-05	1.09E-05	2.98E-04	1.09E-05	1.09E-05
POR-C	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05	3.19E-04	1.09E-05	1.09E-05
SAV-D	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06	1.13E-05	6.22E-06	6.22E-06
SAV-C	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05	1.30E-05	1.14E-05	1.14E-05
SEA-C	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04	1.36E-04	6.79E-04	6.79E-04
TAC-D	5.33E-06	1.08E-05	1.08E-05	5.33E-06	1.08E-05	1.08E-05	5.33E-06	1.14E-05	1.14E-05
TAC-C	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05	1.08E-05
WIL-D	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04	3.26E-04	1.16E-04	1.16E-04
WIL-C	3.04E-04	1.27E-05	1.27E-05	3.04E-04	1.08E-04	1.08E-04	3.04E-04	1.08E-04	1.08E-04
CHN-D	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06	6.27E-06	3.89E-06	3.89E-06

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

Table D-34 Probability of Peak Results, Variable Meteorology (Continued)

Individual Center-line EDE Whole Body Dose, 0-1.6 KM (SV)

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CHA-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CNC-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
CNC-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
GAL-D	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
GAL-C	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
JAC-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
JAC-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
LOS-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
LOS-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
MOT-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NEW-D	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NEW-C	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NOR-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NOR-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
PHI-D	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
PHI-C	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
POR-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
POR-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
SAV-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SAV-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SEA-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
WIL-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
WIL-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
CHN-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03

Individual Center-line Cancer Risk, 0-1.6 KM

	BR-2			RHF			TRIGA		
Site/Loc	EA4	EA5	EA6	EA4	EA5	EA6	EA4	EA5	EA6
CHA-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CHA-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
CNC-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
CNC-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
GAL-D	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
GAL-C	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03	3.73E-02	1.57E-03	1.57E-03
JAC-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
JAC-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
LOS-D	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
LOS-C	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04	1.22E-01	2.00E-04	2.00E-04
MOT-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NEW-D	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NEW-C	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03	3.45E-03	1.06E-03	1.06E-03
NOR-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
NOR-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
PHI-D	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
PHI-C	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04	7.89E-02	5.14E-04	5.14E-04
POR-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
POR-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
SAV-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SAV-C	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03
SEA-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-D	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
TAC-C	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04	2.91E-02	4.00E-04	4.00E-04
WIL-D	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
WIL-C	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03	8.16E-03	1.46E-03	1.46E-03
CHN-D	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03	2.36E-02	1.66E-03	1.66E-03

CHA = Charleston (Wando Terminal), SC; CNC = Concord, CA; GAL = Galveston, TX; JAC = Jacksonville, FL; LOS = Long Beach, CA; MOT = MOTSU, SC; NEW = Elizabeth, NJ; NOR = Norfolk, VA; PHI = Philadelphia, PA; POR = Portland, OR; SAV = Savannah, GA; SEA = Seattle, WA; TAC = Tacoma, WA; WIL = Wilmington, NC; CHN = NWS Charleston, SC

population within 80.5 km (50 mi) of the Elizabeth channel accident location is about 16 million people and typical plumes are about two compass sectors wide, a typical accident plume might expose about two million people to radiation. Thus, for the largest mean result obtained, an average 50-year individual dose over the total exposed populations is about 6,900 person-rem/2,000,000 people = 0.0035 rem per person, which is 5,300 times smaller than the average dose (15 rem) people normally receive from natural, medical, and occupational exposures during the same period of time (BEIR, 1990).

Due to variable weather conditions, the calculated accident consequences vary over a range of values of approximately two orders of magnitude. Quantile values are one means used to indicate how much variation exists among the quantified consequences. The 99.9th quantile values presented in Table D-32 represent the accident consequences that are expected no more than 0.1 percent of the time, that is 99.9 percent of the time the accident consequences will be less than the values presented here. The 99.9th quantile values range from 0.00625 rem (at the MOTSU dock, TRIGA fuel, release category 4) to 108,000 rem (at the Elizabeth channel, BR-2 fuel, release category 6). These results are about three orders of magnitude less likely than the mean, but are less than two orders of magnitude higher than the mean results. (In some cases a 99.9th quantile value is listed as "NOT FOUND." In these instances the peak values, discussed in the following paragraph, occur with a probability of greater than 0.001).

Table D-33 shows that the largest value (peak result) calculated for population dose within 80.5 km (50 mi) of the accident location was 145,000 person-rem (1,450 person-Sv) and that this result was obtained for the Elizabeth channel calculation that used the BR-2 inventory, severity category 6 (EA6) release fractions, and New York City weather. Dividing by the two million people exposed by the accident gives an average 50-year individual dose over the exposed population of about 73 mrem, which is still 250 times smaller than a normal annual individual dose from background and medical exposure over the same period of time. In addition, Table D-34 shows that the probability of this result was 0.0000765 conditional on the accident having occurred. Since the probability of this accident occurring is about 6×10^{-10} per port call, the chance of having this result is much less than 1×10^{-10} per port call.

Table D-31 also shows that mean (expected) 50-year individual centerline doses at a distance of 0.8 km (0.5 mi) from the accident location [the midpoint of the 0-1.6 km or 0-1 mi computational interval] range from a low of 0.000006 rem (0.00000006 Sv) for the Norfolk and MOTSU calculations that used the TRIGA inventory, severity category 4 (EA4) release fractions, and Cape Hatteras weather to a high of 117 mrem (0.00117 Sv) for the Elizabeth calculations that used the BR-2 inventory, severity category 5 (EA5) release fractions, and New York City weather. Thus, the largest expected individual dose is 190 times smaller than a normal background medical and occupational individual dose during the same period (50 years), which suggests that the mean risk to a maximally exposed member of the general population is not of great concern. Note that the channel and dock values for centerline doses are the same for each port. This is because MACCS, in calculating centerline doses, develops the dose for a hypothetical person and so does not take into account population distribution. Therefore, the usually minor difference in position between the dock and channel does not result in different values. Table D-33 shows that the largest value (peak result) calculated for 50-year individual centerline dose for a person located 0.8 km (0.5 mi) from the accident location was 6.6 rem (0.066 Sv) and that this result was obtained for the Philadelphia calculations that used the BR-2 inventory, severity category 6 (EA6) release fractions, and Washington, DC, weather. This dose of 6.6 rem is less than half of the dose received due to background radiation over the same 50-year period. Table D-34 shows that the probability of this result is 0.00051 conditional on the accident having occurred. Thus, the chance per port call of the MEI receiving this 50-year dose is significantly less than 1×10^{-10}.

Table D-31 shows that the mean number of cancer deaths predicted to occur during the decades after the accident, among the populations located within 80.5 km (50 mi) of the accident site at the time of the accident, ranges from 0.00000041 for the MOTSU dock calculation that used the TRIGA inventory, severity category 4 (EA4) release fractions, and mean Cape Hatteras weather to 2.9 for the Elizabeth channel calculation that used the BR-2 inventory, severity category 5 (EA5) release fractions, and mean New York City weather. If all three of the cancer deaths predicted to occur as a result of the accident at the Elizabeth site should happen to occur in the same year, then the death rate among the two million people exposed to radiation by this accident would be 3/2,000,000 = 0.0000015 deaths per person year. Since the normal death rate due to all types of cancer is about 150 deaths per 100,000 people per year (World Almanac, 1992) or 0.0015 deaths per person year, the largest mean (expected) death rate for any base case calculation is 1,000 times smaller than the normal death rate due to cancer. Table D-33 shows that the largest number of cancer deaths obtained for any weather trial in any base case calculation was 60 and that this result was obtained for the Elizabeth channel calculation that used the BR-2 inventory, severity category 6 (EA6) release fractions, and New York City weather. Again, if all of these deaths were to occur in the same year in the future (a very improbable outcome), the death rate during that year among the population exposed to radiation by the accident would be 0.00003 or 50 times lower than the normal death rate due to cancer among this population. Table D-34 shows that the probability of this result is 0.000077 conditional on the occurrence of the accident or less than 1×10^{-10} per port call. Thus, even the worst case number of cancer deaths would be wholly undetectable in the exposed population by the best of epidemiological studies.

Figures D-56 and D-57 present Complementary Cumulative Distribution Functions for population dose and cancer fatalities among the population located within 80.5 km (50 mi) of the accident site for seven of the thirteen ports studied. Only seven were plotted to simplify the figure; these seven provide the full range of results. The figures display the range and probability (conditional on the occurrence of the accident) of these two consequence measures. Figure D-56 shows that any large accident (severity category 5 with the BR-2 inventory is a severe ship collision and fire accident) will lead to a population dose of 10 person-rem, that the values of the 99th quantile (probability of 0.01) range from about 2,000 person-rem to about 40,000 person-rem, and that the largest (peak) result calculated ranges from about 4,600 rem (MOTSU) to about 110,000 rem (Elizabeth). Figure D-57 shows that a large accident has about one chance in 10 (range of 0.002 to 0.6) of causing at least one cancer death among the exposed population in future years, that the values of the 99.9th quantile range from 1 cancer fatality to about 25 cancer deaths, and that the largest (peak) result calculated ranges from 2.1 to 47 deaths due to cancer during the years after the accident.

Figure D-58 presents an example of Complementary Cumulative Distribution Functions for population dose and cancer fatalities for the distance range 0 to 80.5 km (0 to 50 mi) for both the dock and channel locations at Charleston. This figure shows that the dock and channel Complementary Cumulative Distribution Functions for both population dose and cancer fatalities are quite similar, which is typical for all of the ports examined. This suggests that moving the coordinates of the origin of a population distribution a small distance (a few kilometers) has little effect on population dose or cancer fatalities among population located within 80.5 km (50 mi) of the accident location for severe accidents (Table D-28 lists the coordinates of the origins of the polar coordinate population distributions used in these calculations).

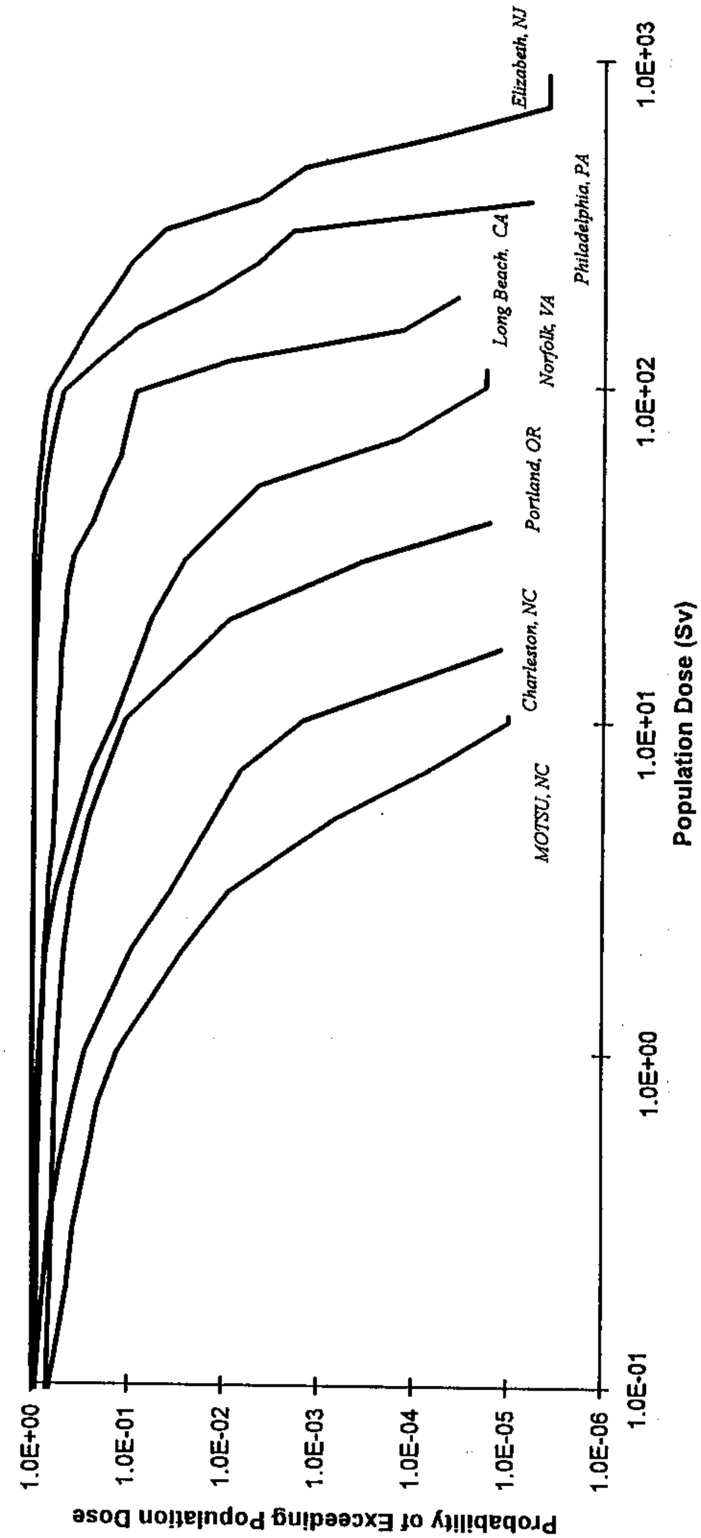


Figure D-56 Effective Dose Equivalent Whole Body Population Dose, 0-80 km (0-50 mi), Select Ports (at the Dock), Variable Meteorology, BR-2 Inventory, Severity Category 5 Releases

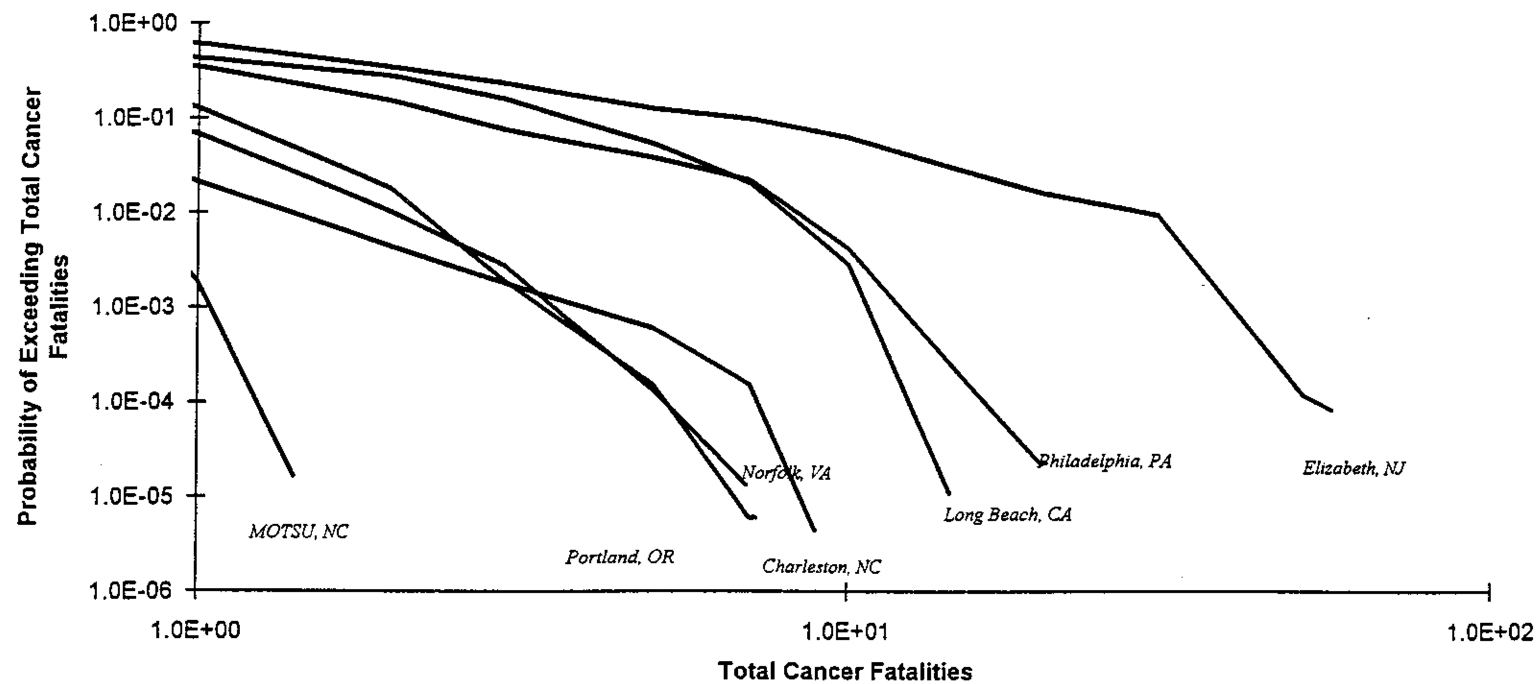


Figure D-57 Total Latent Cancer Fatalities, 0-80 km (0-50 mi), Select Ports (in the Channel), Variable Meteorology, BR-2 Inventory, Severity Category 5 Release

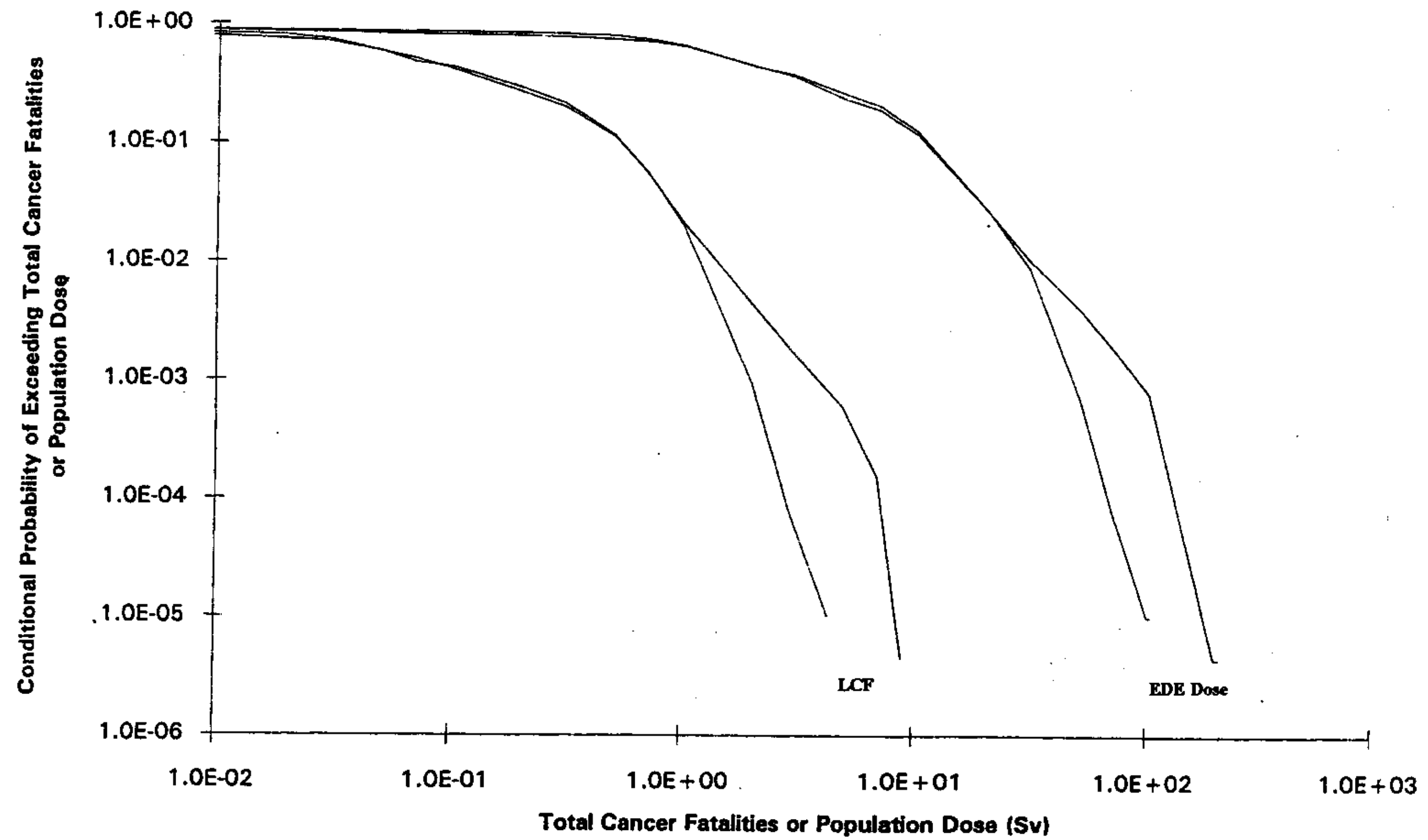


Figure D-58 Effective Equivalent Dose Whole Body Population Dose (Person-Sv) and Total Latent Cancer Fatalities, 0-80 km (0-50 mi), Charleston Dock and Channel Locations, Variable Meteorology, BR-2 Inventory, Severity Category 5 Release

D.5.4.3 Sensitivity Calculations

Two principal sensitivity calculations were performed to determine the sensitivity of the results to key parameters. First, the effect of using local less detailed meteorological data versus meteorological data recorded at a National Weather Service station located some distance from the port was evaluated. Second, the results of exceptionally high spent nuclear fuel temperatures were examined. Additionally, the sensitivity of changes in plume buoyancy, the size of the nuclide set, modal study release fractions, corrosion products release, and work force population were examined. The meteorological sensitivity calculations compared results obtained using variable meteorology recorded at a National Weather Service station away from the port to results obtained using constant meteorology recorded at the port. All other sensitivity calculations except the work force calculations were performed by modifying the Elizabeth base case channel calculation as was appropriate in order to examine the parameter of interest. The work force calculations were based on the Elizabeth dock site. All of the sensitivity calculations used the BR-2 inventory and all, except those that examined release fractions, used severity category 5 release fractions.

D.5.4.3.1 Variable vs. Constant Meteorology

Variable meteorology, which takes into account hourly changes of wind direction and speed, was used in the calculations that led to the results presented in this EIS. However, the detailed weather data required to support these calculations are not available in most ports, so detailed data from the most appropriate National Weather Service Station location possible were used. A sensitivity study was performed to better understand the effect of using detailed but not local weather data versus using local less-detailed port weather data. The local weather is called constant meteorology, to reflect the fact that the weather remains constant during the course of the accident, not varying on an hourly basis.

This study performed, for each port, a large number of constant meteorology calculations for each port, using the conditions and probabilities specified in the joint frequency distributions that were available for each port. Since joint frequency distributions specify for each compass sector the probability of occurrence of each of the six Pasquil-Gifford atmospheric stability classes with each of six windspeed ranges, 16 x 6 x 6 = 576 constant meteorology calculations could be performed, once assuming that it was raining and once assuming that it was not. Then, by cumulating the results of each set of approximately 1,150 constant meteorology single weather trial calculations (rain does not occur for all of the sets of conditions in the joint frequency distribution), a Complementary Cumulative Distribution Function could be constructed to compare with the complementary cumulative distribution function obtained using variable meteorology recorded at the nearest National Weather Service Station.

Table D-35 presents a sample joint frequency table for one of the ports examined during this EIS (Charleston). Tables D-36 and D-37 present the port wind rose and a probability of rain by stability class respectively for selected ports.

Constant meteorology calculations were performed as follows. For each port examined, two sets of constant meteorology calculations were performed. Both used the joint frequency distribution of windspeed (6 windspeed ranges) and stability class (6 stability classes) by wind direction (16 compass sectors) for the port being analyzed as the meteorological input data for MACCS. Each calculation was run two times, once for no rain and once assuming that it was raining throughout the entire simulation. Therefore, each MACCS constant meteorology calculation consisted of 6 x 6 x 16 x 2 = 1,152 constant meteorology trials. From these 1,152 trials, a complementary cumulative Distribution Function and a mean (expected result) was constructed for each consequence measure calculated.

Table D-35 1988-92 Summary Joint Frequency Table for Charleston, SC Port

A Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0003	.0002	.0002	.0002	.0002	.0002	.0002	.0002	.0003	.0003	.0002	.0003	.0001	.0002	.0001	.0001
4- 7	.0005	.0003	.0005	.0004	.0004	.0006	.0003	.0003	.0006	.0004	.0004	.0002	.0002	.0001	.0002	.0003
8-12	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

B Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0008	.0003	.0006	.0007	.0005	.0007	.0008	.0005	.0011	.0006	.0010	.0007	.0007	.0006	.0004	.0005
4- 7	.0018	.0013	.0017	.0019	.0023	.0020	.0017	.0013	.0031	.0016	.0023	.0017	.0018	.0011	.0013	.0007
8-12	.0021	.0013	.0021	.0025	.0025	.0014	.0013	.0008	.0016	.0013	.0013	.0010	.0013	.0012	.0016	.0012
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

C Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0002	.0002	.0003	.0003	.0003	.0003	.0003	.0003	.0004	.0003	.0005	.0005	.0003	.0002	.0001	.0001
4- 7	.0015	.0014	.0016	.0022	.0026	.0021	.0019	.0020	.0026	.0021	.0035	.0021	.0014	.0010	.0014	.0012
8-12	.0081	.0038	.0061	.0072	.0090	.0049	.0037	.0031	.0057	.0051	.0062	.0042	.0034	.0043	.0042	.0049
13-18	.0017	.0012	.0021	.0020	.0021	.0014	.0006	.0005	.0006	.0005	.0005	.0004	.0008	.0005	.0008	.0007
19-24	.0000	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

D Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0009	.0005	.0005	.0007	.0006	.0005	.0004	.0008	.0013	.0012	.0013	.0012	.0007	.0007	.0004	.0004
4- 7	.0045	.0030	.0037	.0027	.0028	.0026	.0022	.0044	.0094	.0079	.0093	.0056	.0042	.0039	.0026	.0022
8-12	.0196	.0151	.0165	.0112	.0108	.0070	.0047	.0061	.0168	.0216	.0201	.0124	.0108	.0075	.0066	.0083
13-18	.0140	.0179	.0145	.0082	.0133	.0096	.0058	.0058	.0084	.0080	.0057	.0047	.0051	.0035	.0034	.0036
19-24	.0009	.0019	.0020	.0007	.0020	.0023	.0010	.0004	.0003	.0000	.0000	.0001	.0001	.0001	.0001	.0000
>24	.0002	.0005	.0002	.0001	.0003	.0003	.0001	.0000	.0000	.0001	.0000	.0000	.0000	.0000	.0000	.0000

E Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
4- 7	.0127	.0071	.0088	.0047	.0033	.0023	.0021	.0025	.0050	.0061	.0094	.0066	.0052	.0049	.0036	.0051
8-12	.0063	.0072	.0085	.0059	.0067	.0058	.0032	.0030	.0038	.0066	.0043	.0021	.0019	.0013	.0011	.0016
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

F Stability

Wind Speed (mph)	N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
	Wind Directions (Blowing Toward)															
1- 3	.0090	.0076	.0081	.0059	.0052	.0039	.0035	.0057	.0086	.0076	.0101	.0054	.0054	.0036	.0032	.0035
4- 7	.0122	.0088	.0112	.0074	.0063	.0051	.0044	.0059	.0095	.0104	.0122	.0057	.0044	.0044	.0029	.0039
8-12	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
13-18	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
19-24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000
>24	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000	.0000

Table D-36 Wind Rose Table for Select Ports

1988-92 Summary Wind Rose Table For Charleston, SC Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0974	.0796	.0892	.0650	.0712	.0530	.0380	.0436	.0790	.0817	.0882	.0549	.0479	.0389	.0341	.0385

1988-92 Summary Wind Rose Table For Long Beach, CA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0246	.0171	.0602	.3093	.1804	.0157	.0177	.0229	.0331	.0227	.0271	.0475	.1115	.0601	.0348	.0154

1988-92 Summary Wind Rose Table For Newark, NJ Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0784	.0725	.1015	.0871	.0854	.0639	.0788	.0559	.0832	.0786	.0442	.0273	.0231	.0304	.0452	.0447

1988-92 Summary Wind Rose Table For Norfolk, VA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.1078	.0963	.1021	.0647	.0562	.0456	.0344	.0285	.0940	.0665	.0860	.0573	.0470	.0321	.0358	.0458

1988-92 Summary Wind Rose Table For Philadelphia, PA Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0682	.0440	.0950	.1118	.1281	.0913	.0715	.0568	.0669	.0266	.0275	.0639	.0545	.0284	.0278	.0378

1988-92 Summary Wind Rose Table For Portland, OR Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0936	.0551	.0337	.0315	.0756	.0956	.1163	.1054	.0704	.0187	.0171	.0225	.0638	.1126	.0576	.0304

1988-92 Summary Wind Rose Table For Wilmington, NC Port
Wind Directions (Blowing Toward)

N	NNE	NE	ENE	E	ESE	SE	SSE	S	SSW	SW	WSW	W	WNW	NW	NNW
.0744	.0804	.0994	.0798	.0747	.0378	.0417	.0435	.0955	.0780	.0699	.0488	.0549	.0351	.0411	.0451

Table D-37 Rainfall Data, Select Ports

Rainfall Data for the Charleston, SC Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.05000	0.00264
B	0.22400	0.00371
C	0.16322	0.00771
D	0.13860	0.11099
E	0.14740	0.01099
F	0.07941	0.00125

Rainfall Data for the Long Beach, CA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.00000	0.00000
C	0.14375	0.00233
D	0.07837	0.03648
E	0.06596	0.00809
F	0.07083	0.00115

Rainfall Data for the Newark, NJ Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.05000	0.00059
C	0.08571	0.00648
D	0.08577	0.12139
E	0.08968	0.00971
F	0.05000	0.00153

Rainfall Data for the Norfolk, VA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.09167	0.00371
C	0.10921	0.00771
D	0.47136	0.11099
E	0.12574	0.01099
F	0.05000	0.00125

Rainfall Data for the Philadelphia, PA Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.17500	0.00089
C	0.11250	0.00431
D	0.07520	0.12101
E	0.10682	0.00649
F	0.17500	0.00035

Rainfall Data for the Portland, OR Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.11250	0.00139
C	0.08125	0.01245
D	0.06172	0.15220
E	0.06493	0.01428
F	0.05000	0.00087

Rainfall Data for the Wilmington, NC Port 1988-1992 Data

Stab Class	Avg Rate (in/hr)	Fraction Time
A	0.00000	0.00000
B	0.18235	0.00718
C	0.17500	0.01937
D	0.15048	0.12490
E	0.16295	0.02310
F	0.08571	0.00244

Table D-38 Comparison of Population Dose and Selected Ports Using Variable vs. Constant Meteorology for Category Accident of a BR-2 Fuel Cask

Site/Loc	EDE Whole Body Population Dose, 0-80 KM (Sv) Mean		99.9th Quantile		Total Cancer Fatalities, 0-80 KM Mean		99.9th Quantile	
	Var	Const	Var	Const	Var	Const	Var	Const
CHA-D	4.15E+00	3.06E+00	4.63E+01	2.13E+01	1.89E-01	1.29E-01	1.98E+00	8.43E-01
CHA-C	4.18E+00	3.41E+00	9.03E+01	3.71E+01	1.90E-01	1.43E-01	3.96E+00	2.01E+00
LOS-D	4.71E+01	3.44E+01	2.67E+02	1.19E+02	1.99E+00	1.44E+00	1.03E+01	5.35E+00
LOS-C	4.26E+01	3.31E+01	2.19E+02	8.16E+01	1.80E+00	1.38E+00	9.81E+00	3.43E+00
NEW-D	6.55E+01	5.47E+01	5.87E+02	2.32E+02	2.75E+00	2.28E+00	2.46E+01	9.54E+00
NEW-C	6.93E+01	5.89E+01	9.41E+02	NOT-FOUND	2.90E+00	2.46E+00	3.89E+01	NOT-FOUND
NOR-D	8.54E+00	8.88E+00	1.03E+02	7.26E+01	3.77E-01	3.72E-01	4.23E+00	3.07E+00
NOR-C	6.65E+00	6.76E+00	9.02E+01	3.51E+01	2.96E-01	2.83E-01	3.59E+00	1.34E+00
PHI-D	2.81E+01	2.53E+01	3.10E+02	NOT-FOUND	1.20E+00	1.06E+00	1.18E+01	1.34E+00
PHI-C	2.74E+01	2.01E+01	2.86E+02	5.91E+01	1.17E+00	8.40E-01	1.22E+01	2.45E+00
POR-D	1.17E+01	1.08E+01	1.09E+02	7.76E+01	5.18E-01	4.54E-01	4.90E+00	3.29E+00
POR-C	1.12E+01	8.76E+00	1.01E+02	3.88E+01	4.97E-01	3.68E-01	3.78E+00	1.51E+00
MOT-D	2.08E+00	1.02E+00	2.46E+01	NOT-FOUND	9.94E-02	4.37E-02	1.22E+00	1.51E+00
WIL-C	2.07E+00	1.05E+00	2.25E+01	5.47E+00	9.76E-02	4.49E-02	1.04E+00	2.22E-01

Table D-38 compares for seven ports the expected (mean) and 99.9th quantile values of population dose and cancer fatalities for the distance range 0 to 80.5 km (0-50 mi) obtained using variable meteorology to the values obtained using constant meteorology. Inspection of the table shows that the mean values for constant meteorology are quite similar to mean values for variable meteorology. For example, for population dose, the ratio of the variable meteorology result to the constant meteorology result has an average value and standard deviation of 1.31±0.31 for population dose and 1.34±0.41 for cancer fatalities. The MOTSU dock calculation yielded the largest values for these ratios, 2.04 for mean population dose and 2.27 for cancer fatalities. Thus, the use of meteorological data recorded at a nearby National Weather Service station yields expected (mean) values for population dose and cancer fatalities that are on average about 30 to 40 percent larger than the values obtained using constant meteorological conditions for each of the six Pasquil-Gifford atmospheric stability classes that were derived from data recorded at the harbor.

The 99.9th quantile values of population dose and cancer fatalities among the population that resides within 80.5 km (50 mi) of the harbor results agree less well for constant and variable meteorology. For several ports, the 99.9th quantile population dose is missing, ("not found"), for the constant meteorology calculation. This means that the probability of the largest result obtained for any of the 1,152 trials run during each constant meteorology calculation was larger than 0.001 for that particular calculation. For the locations that yielded a 99.9th quantile value for both the variable and constant meteorology calculation, the ratio of the 99.9th quantile variable meteorology result to the 99.9th quantile constant meteorology result has a value of 2.64±0.98 for population dose and 3.00±2.02 for cancer fatalities. The fact that the 99.9th quantile values obtained using variable meteorology are on average 2.5 to 3.0 times larger than the 99.9th quantile values obtained using constant meteorology suggests that the importance sampling scheme, used by MACCS to select weather sequences from a year of variable meteorological data, is able to find weather sequences that lead to adverse results that are not represented in the sets of constant meteorological conditions found in the joint frequency distributions of windspeed and atmospheric stability by wind direction that were recorded at the harbors. This is so because the occurrence of rain is usually the cause of peak results at some later time when the plume is passing over some downwind highly populated region. Thus, because rain at some downwind location was not modeled by the constant meteorology calculations, these results should differ significantly from those obtained using variable meteorology, especially for the higher quantiles of result distributions.

Figure D-59 presents, as an example, complementary cumulative distribution functions for Long Beach of the 50-year population dose and lifetime LCFs over the distance range from 0 to 80.5 km (0 to 50 mi) obtained using both variable and constant meteorology. All four calculations used the BR-2 inventory and severity category 5 release fractions. The dose calculation was performed for the dock location at Long Beach; and the LCF calculation was performed for the channel location. Inspection of the figures shows that the constant and variable meteorology complementary cumulative distribution functions are quite similar until the 90th quantile of the distributions are reached, and diverge increasingly as higher quantiles are passed, with the constant meteorology complementary cumulative distribution function generally falling off faster than the variable meteorology complementary cumulative distribution function (smaller consequence value at any consequence probability). Thus, the figures confirm the conclusion reached by inspection of Table D-38, that variable and constant meteorology yield quite similar estimates for mean results and that adverse meteorological conditions are more likely to be modeled if weather sequences are selected by importance sampling from a year of variable data than if constant meteorological conditions are used.

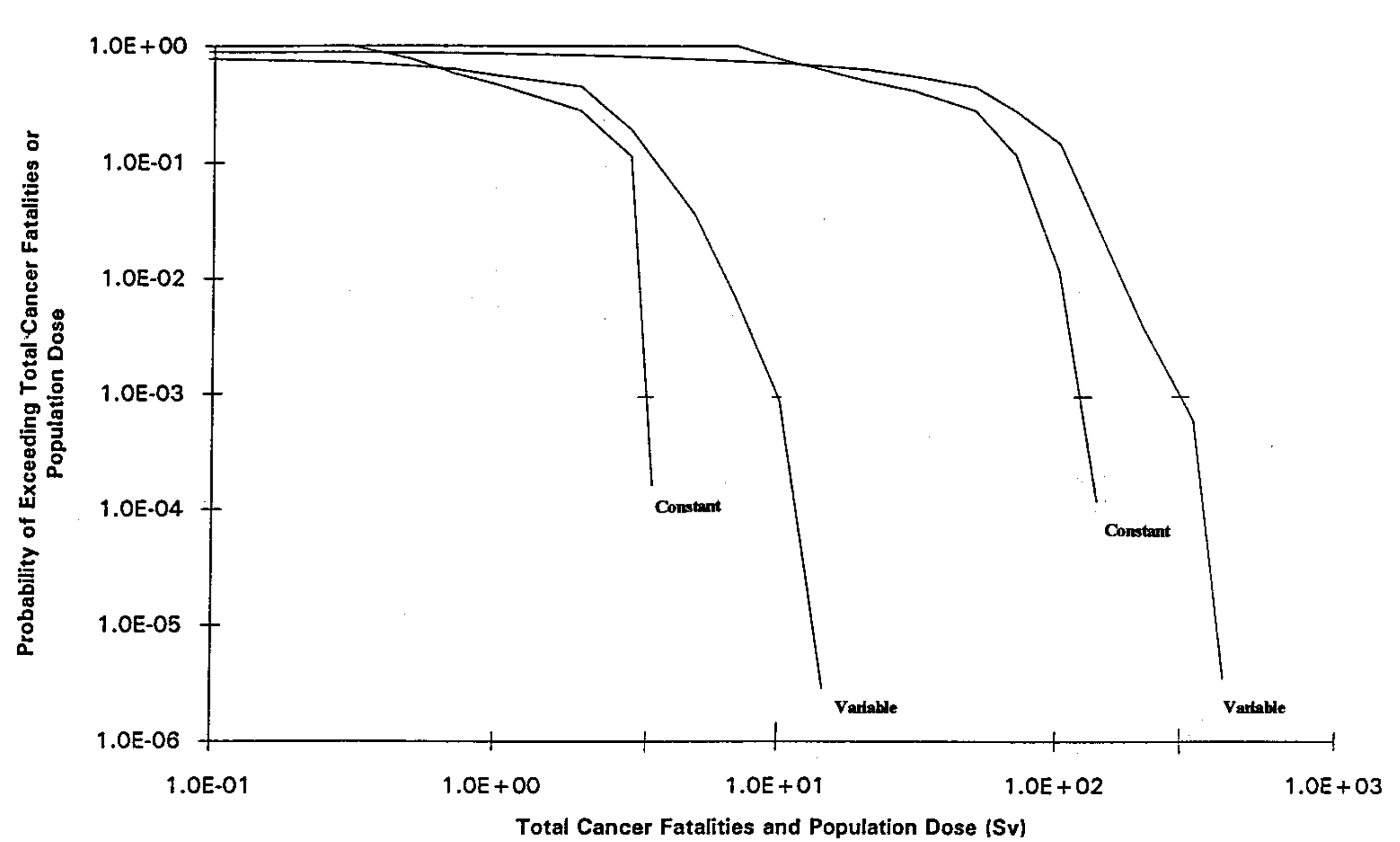


Figure D-59 Effective Dose Equivalent Whole Body Population Dose (Sv) (Dock) and Total Cancer Fatalities (Channel), 0-80 km (0-50 mi), Long Beach, Variable and Constant Meteorology, BR-2 Inventory, Severity Category 5 Releases

D.5.4.3.2 High-Temperature Sensitivity Calculations

As previously discussed, releases of radioactive material from spent nuclear fuel transportation casks are categorized by severity. Severity category 6, which results in the largest release, is assumed for the marine transportation portion of this EIS to be caused by a severe ship collision that results in damage to the transportation cask and a severe fire that engulfs the cask. Only around one in five severe ship fires reach temperatures above approximately 700°K or 800°F (see Attachment D5, Section 4). As discussed below, extremely high temperatures, above 900°K (1,160°F), result in phenomena that could significantly alter the release fraction for aluminum-based and TRIGA fuel (previous studies have not specifically addressed the impact of these phenomena). Therefore, the release fractions assumed for severity category 6 (Table D-21) are for temperatures of the spent nuclear fuel above 700°K (800°F) but below 900°K (1,160°F).

Section D.5.3.1 of this appendix developed probabilities of the more severe marine accidents. Table D-24 stated that the probability of a severity category 6 accident is 6×10^{-10}, or less than one chance in a billion per cask shipment. This very low probability is made even lower if the probability of the severe fire causing the spent nuclear fuel temperature to exceed 900°K (1,160°F) is considered. Appendix D Attachment D5 concludes that the probability of a severe ship fire exceeding spent nuclear fuel temperatures of 900°K (1,160°F) is 0.1. Multiplying the probability of a severity category 6 accident (6×10^{-10}) by the probability of a severe fire on the ship (0.1) results in the probability of a severity category 6 accident that includes a severe ship fire, 6×10^{-11}. This exceedingly small probability indicates that the occurrence of this condition is not a creditable accident. However, for completeness, an evaluation of the consequences of such an accident is presented below as a sensitivity calculation.

The review of the behavior of aluminum-uranium (Al-U) alloy and TRIGA fuels at temperatures above 900°K (1,160°F), presented in Attachment D5, found that at these temperatures Al-U fuels melt, and if exposed to air, TRIGA fuel burns. Table D5-2 (in Attachment D5) compares the release fractions estimated for these high-temperature scenarios to those used in the base case calculations. These data show the high-temperature events (the category 5B and 6B events) increase release from these fuels significantly.

Since both processes (melting and burning) are expected to produce fission product release fractions that are significantly larger than those used during base case calculations for severity category 6 accidents, sensitivity calculations were performed so that the consequences and risks associated with these larger releases could be compared to the consequences and risks of the base case results. Again the Elizabeth channel location was used to perform the sensitivity calculations. Three calculations were performed, two BR-2 aluminum-uranium alloy fuel calculations and one TRIGA fuel calculation. All of these calculations used the release fractions specified in Table D-39 for high-temperature scenarios. The first aluminum-uranium alloy fuel sensitivity calculation used severity category 5B and the second category 6B release fractions. The single TRIGA sensitivity calculation used category 6B release fractions. Calculations were not performed with any of the other sets of release fractions presented in Table D-39, because each of the other sets is smaller than the set used in the base case calculations that it would replace; and would thus yield smaller consequences and risks.

Table D-39 presents the results of these high-temperature sensitivity calculations and compares them to the base case results obtained using the same inventories but using the severity category 5 or 6 release fractions given in Table D-21. Table D-39 shows that, as expected, the larger severity category 5B and severity category 6B release fractions lead to consequences significantly larger than those obtained for the base case calculations that used severity category 5 and severity category 6 release fractions. Inspection of the table shows that the larger release fractions increase consequence estimates by factors of ten to 100.

Table D-39 High-Temperature Sensitivity Calculation Results

	BR-2				*TRIGA*	
Accident Severity Category	5	5B	6	6B	6	6B
Accident Probability	5×10^{-9}	5×10^{-10}	6×10^{-10}	6×10^{-11}	6×10^{-10}	6×10^{-11}
Peak Probability [0-1.6 km [0-1 mi)]	8.41×10^{-5}	7.65×10^{-5}	8.41×10^{-5}	7.03×10^{-5}	8.41×10^{-5}	2.17×10^{-4}
Peak Probability [0.80.5 km (0-50 mi)]	8.41×10^{-5}	1.16×10^{-5}	8.41×10^{-5}	1.45×10^{-5}	8.41×10^{-5}	1.45×10^{-5}
			EDE Whole Body Population Dose (person-rem)			
0-1.6 km (0-1 mi)						
Mean	236	1,490	192	3,810	26.8	3,980
Peak	42,100	203,000	45,900	271,000	6,390	297,000
0-80.5 km (0-50 mi)						
Mean	6,930	68,400	6,770	639,000	937	298,000
Peak	133,000	1,450,000	145,000	14,400,000	20,200	6,390,000
			Total Cancer Fatalities			
0-1.6 km (0-1 mi)						
Mean	0.098	0.622	0.0802	1.59	0.0112	1.66
Peak	17.5	84.5	19.1	113	2.66	123
0-80.5 km (0-50 mi)						
Mean	2.90	28.7	2.84	268	0.392	125
Peak	55.3	603	60.4	6,000	8.39	2,660
			Impact Distances (km)			
Decontamination						
Mean	0.0	0.0156	0.0	0.302	0.0	0.0993
Peak	0.0	1.61	0.0	8.05	0.0	6.44
Cond. Peak Prob.	---	0.00969	---	0.00116	---	7.53×10^{-5}
Interdiction						
Mean	0.0	0.0156	0.0	0.302	0.0	0.0993
Peak	0.0	1.61	0.0	8.05	0.0	6.44
Cond. Peak Prob.	---	0.00969	---	0.00116	---	7.53×10^{-5}
Condemnation						
Mean	0.0	0.0	0.0	0.0292	0.0	0.00263
Peak	0.0	0.0	0.0	3.22	0.0	1.61
Cond. Peak Prob.	---	---	---	0.000648	---	0.00163
			Population Dose Risk			
0-1.6 km (0-1 mi)						
Mean	1.2×10^{-6}	7.5×10^{-7}	1.2×10^{-7}	2.3×10^{-7}	1.6×10^{-8}	2.4×10^{-7}
Peak	1.8×10^{-8}	7.8×10^{-9}	2.3×10^{-9}	1.1×10^{-9}	3.2×10^{-10}	3.9×10^{-9}
0-80.5 km (0-50 mi)						
Mean	3.5×10^{-5}	3.4×10^{-6}	4.1×10^{-6}	3.8×10^{-5}	5.6×10^{-7}	1.8×10^{-5}
Peak	5.6×10^{-8}	8.4×10^{-9}	7.3×10^{-9}	1.3×10^{-8}	1.0×10^{-9}	5.5×10^{-9}
			Cancer Fatality Risk			
0-1.6 km						
Mean	4.9×10^{-10}	4.4×10^{-10}	4.8×10^{-11}	9.5×10^{-11}	6.7×10^{-12}	1.0×10^{-10}
Peak	7.4×10^{-12}	3.2×10^{-12}	9.6×10^{-13}	4.8×10^{-13}	1.3×10^{-13}	1.6×10^{-12}
0-80.5 km						
Mean	1.5×10^{-7}	1.6×10^{-7}	1.7×10^{-9}	1.6×10^{-8}	2.4×10^{-10}	7.5×10^{-9}
Peak	2.3×10^{-11}	3.5×10^{-12}	3.0×10^{-12}	5.2×10^{-12}	4.2×10^{-13}	2.3×10^{-12}

However, because the probabilities of occurrence of these high-temperature release fractions (see Attachment D-5) for BR-2 aluminum uranium alloy fuel inventories are generally ten times smaller than those associated with the severity category 5 and severity category 6 accident categories, the risks associated with these larger releases are comparable to or smaller than those predicted for base case BR-2 calculations. For TRIGA fuel, severity category 6B release fractions are much larger than the severity category 6 release fractions. The probability of the severity category 6B release fractions is only ten times smaller than that of the severity category 6 release fractions. Therefore, the risks associated with a TRIGA fuel category 6B release are significantly larger than those obtained for the base case accident severity category 6 calculation. But, because the TRIGA inventory is substantially smaller than the BR-2 inventory, the TRIGA severity category 6B risks are still smaller than the risks obtained for base case calculations using the BR-2 inventory and the severity category 5 set of release fractions.

Other environmental impacts in addition to the public health consequences are presented in Table D-39. These impacts were determined as part of the MACCS calculations. MACCS calculated land impacts based on a habitability dose criterion and cost effectiveness of mitigative actions such as evacuation, temporary relocation, and land decontamination and interdiction. The habitability criterion is based on the need to take action to ensure that the dose to a person remains below 4 rem[1] over a 5-year period. MACCS code determines the mitigative actions in a predetermined sequence in order to select the least stringent action which will allow the habitability dose criterion to be satisfied. The order of actions is: 1) decontamination alone (minimum decontamination process, three levels of decontamination process can be specified), 2) maximum level of decontamination followed by an interdiction period, and 3) permanent interdiction (condemnation) of the land. The decontamination distance is that distance from the accident location that requires post-accident clean-up to ensure this dose level is not achieved. The land is usable, that is, people may live and work in the area, within a relatively short period after the accident. The interdiction distance is that distance from the accident that even after decontamination would require some time, typically seven years, before the land area would be useable. The condemnation distance characterizes the land area that even after decontamination would remain unusable for at least 30 years.

MACCS code calculates both the affected population in the urban areas and the affected farmlands in the rural areas. The affected distances, (i.e., decontamination, interdiction, and condemnation distances), in the rural areas are generally larger than those of the urban area. Since one of the principal uses of rural land is agricultural, the consumption of contaminated food produced in these areas would result in larger doses to some members of the public.

Table D-39 provides the land impact distances for an accident that occurs in the Port of Elizabeth for the most severe accident severity categories of both the base case calculations (category 5 and 6 for the BR-2 fuel and category 6 for the TRIGA fuel) and for the most severe of the high temperature accident scenarios (categories 5B and 6B for BR-2 fuel and category 6B for the TRIGA fuel). Since the ports are located primarily in urban areas, the impact distances presented are those based on the urban (population) impact calculations. For the base case accident scenarios, MACCS predicted no impact on the usability of the land. However, when temperatures reaching the melting point of the aluminum based fuel and the combustion temperature of the TRIGA fuel are realized, some land-use impacts are calculated. All mean impact distances are well under 1 km (0.6 mi), with the largest distance being approximately 300 m

1 *This arises from 2 rem in first year and 0.5 rem per year for the years 2 to 5. This criterion is consistent with the Environmental Protection Agency's long-term objectives of the Protective Action Guide, (Section 4.2.1 of "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," EPA 1991).*

(1000 ft). The peak values quoted in Table D-39 represent the worst possible consequences, driven by meteorological conditions that create the maximum potential damage. The occurrences of these meteorological conditions are of low probabilities which are ranging from approximately one-in-one hundred to less than one-in-ten thousand.

In addition to the Port of Elizabeth, the land impact analysis was performed for several of the candidate ports, including Concord NWS, CA; Galveston, TX; MOTSU, NC; and Tacoma, WA. For these four ports, the mean values for the land impacts resulting from the category 6B accidents, the most severe of all accident categories, were of the same order of magnitude as, and slightly smaller than, the results presented in Table D-39 for the Port of Elizabeth.

D.5.4.3.3 Other Sensitivity Calculations

In addition to the two sensitivity calculations discussed above, sensitivity calculations were also performed that examined the effect on consequences of (1) plume buoyancy, (2) the size of the set of nuclides used to specify inventories, (3) Modal Study release fractions, (4) corrosion deposits release, and (5) work force population. Table D-40 summarizes the calculations performed. For all of these calculations, the reference calculation was the base case Elizabeth dock or channel calculation that used the BR-2 inventory, severity category 5 release fractions, and variable meteorology recorded at the New York City National Weather Service station. Work force sensitivity calculations used the Elizabeth dock population distribution. All of the other sensitivity calculations used the Elizabeth channel population distribution. Table D-41 presents mean and peak population doses and cancer fatalities for two distance ranges, 0-1.6 km and 0-80.5 km, (0-1 and 0-50 mi) for all of the "other" sensitivity calculations, and also for the reference Elizabeth base case calculations to which sensitivity calculation results should be compared.

D.5.4.3.3.1 Plume Buoyancy

As Table D-21 showed, a severity category 5 release scenario results from a collision and a severe fire. Thus, the first sensitivity calculation performed examined the effect of plume buoyancy (i.e, of plume rise) on accident consequences. This was done by repeating the Elizabeth channel reference calculation setting the sensible heat content of the release to zero. This change produces a cold plume that is not subject to plume rise and thus is not lofted over the population located close to the release point (the accident location). The results of this sensitivity calculation are presented in Table D-41.

Table D-41 shows that changing the reference Elizabeth channel calculation to a cold release not subject to plume rise causes mean and peak population doses and cancer fatalities to increase somewhat for the 0-80.5 km (0-50 mi) distance range and substantially for the 0-1.6 km (0-1 mi) distance range. For the 0-80.5 km (0-50 mi) distance range, mean population dose and cancer fatalities both increase by a factor of 2.4, and peak population dose and cancer fatalities increase by a factor of 1.1. For the 0-1.6 km (0-1 mi) distance range, mean population dose and cancer fatalities both increase by a factor of 17, and peak population dose and cancer fatalities both increase by a factor of 2.7. Thus, if engulfing fires increase release magnitudes, consequence magnitudes will not increase proportionately because the fire will produce a hot plume that will be lofted over nearby populations decreasing radiation exposures and thus health effects among those populations. It should be mentioned that the releases assumed here (category 5) are not considered possible *without* the fire. This calculation was done to show the sensitivity of the results to the presence of a fire.

Table D-40 Other Sensitivity Calculations

Run No.	Meteorology[a]		Nuclides[b]		Release Fractions[c]			Heat[d]		Shielding[e]	
	Variable	*Constant*	*MACCS*	*EIS*	*5*	*MS/nM5*	*MS/M5*	*H*	*C*	*N*	*C*
BC	x			x	x			x		x	
Buoyancy Calculations											
1a.	x			x	x				x	x	
Nuclide Sensitivity Calculations											
2a.	x		x		x			x			x
2b.	x		x		x				x		x
Modal Study Release Fraction Calculations											
3a.	x			x		x		x		x	
3b.	x			x			x	x		x	
Corrosion Products Calculations											
4a.[f]	x			x	EA3				x	x	
4b.[g]	x			x	x			x		x	
Work Force Calculations											
5a.	x			x	x			x		x	
5b.	x			x	x				x	x	
5c.	x			x	x				x		x
5d.[h]	x			x	x				x	x	
5e.[h]	x			x	x				x		x
5f.[i]	x			x	x				x	x	

[a]*Meteorology: Variable = hourly National Weather Service data, Constant = Joint Frequency Data.*

[b]*Nuclides: MACCS = 22 MACCS nuclides, EIS = 34 EIS nuclides.*

[c]*Release Fractions: 5 = severity category 5 release fractions; MS/nM5 = release fractions for nonmetallic (TRIGA) spent nuclear fuel for Modal study cask response region roughly corresponding to severity category 5; MS/M5 = release fraction for metallic (aluminum-based) spent nuclear fuel for Modal study cask response regions roughly corresponding to severity category 5.*

[d]*Heat: H = hot plume, C = cold plume.*

[e]*Shielding: N = normal shielding factors; C = sheltering shielding factors from 0-8 km (0-5 mi) for one day and normal shielding factors at all other times and distances.*

[f]*Only Corrosion Products released*

[g]*With Corrosion Products release added to the reference release.*

[h]*With puff and tail*

[i]*With puff and tail, and evacuation from 0-1.6 km (0-1 mi.)*

D.5.4.3.3.2 Size of Nuclide Set

Table D-25 presented the three inventories used in the base case analyses. Each inventory contains 34 radionuclides. The default set of radionuclides used by MACCS does not contain dose conversion factors for 13 of these 34 radionuclides. These 13 radionuclides are hydrogen-3, tin-123, antimony-125, tellurium-125m, promethium-147, promethium-148m, europium-154, europium-155m, uranium-234, uranium-235, uranium-238, americium-242m, and americium-243. *Chronic* health effect dose conversion factors for all 13 of these radionuclides were available (DOE, 1988a; DOE, 1988b) and were added to the MACCS dose conversion factor library for this study. However, because generally accepted *acute* health effect dose conversion factors were not available, all calculations performed for this study were run not including acute health effects for these 13 radionuclides.

Table D-41 Sensitivity Study Results, Elizabeth Dock and Channel, Inventory BR-2, Severity Category 5

Run	EDE Whole Body Population Dose (person-rem)				Total Cancer Fatalities			
	0-1.6 km (0-1 mi)		0-80.5 km (0-50 mi)		0-1.6 km (0-1 mi)		0-80.5 km (0-50 mi)	
	Mean	Peak	Mean	Peak	Mean	Peak	Mean	Peak
Base Case (Channel)	236	42,100	6,930	133,000	0.099	17.5	2.90	55.3
Buoyancy								
1	4,200	114,000	16,900	151,000	1.75	47.6	7.07	62.9
Nuclide Sensitivity								
2a	236	42,100	6,930	133,000	0.0985	17.5	2.90	55.3
2b	4,200	114,000	16,900	151,000	1.75	47.6	7.07	62.9
Modal Study Release Fraction								
3a	53.7	9,540	1,570	30,100	0.0224	3.98	0.661	12.6
3b	0.3	47.7	7.9	151	0.000112	0.0199	0.00331	0.0628
Corrosion Products Calculations								
4a	739	20,100	2,950	26,600	0.319	8.70	1.27	11.5
4b	278	49,400	8,120	156,000	0.116	20.7	3.42	65.4
Base Case (Dock)	71.3	13,300	6,550	113,000	0.0298	5.56	2.75	47.2
Work Force								
5a	105	14,400	6,600	113,000	0.0438	6.02	2.77	47.2
5b	1,870	40,400	11,200	84,600	0.780	16.8	4.69	35.3
5c	1,860	40,300	11,200	84,500	0.778	16.8	4.68	35.3
5d	1,940	40,400	11,600	72,500	0.810	16.9	4.84	30.2
5e	1,940	40,400	11,500	72,500	0.808	16.8	4.83	30.2
5f	1,940	40,300	11,500	72,500	0.808	16.8	4.83	30.2

The effect of not including acute impacts for 13 of the radionuclides in the inventories was examined by two sensitivity calculations. For these calculations, the reference Elizabeth channel calculation was performed with and without the chronic effects of the 13 radionuclides for two situations, once assuming a hot release, and once assuming a cold release. Table D-41 shows that removing these 13 radionuclides from the BR-2 inventory had no significant impact on either mean or peak values of population dose or cancer fatalities over the distance ranges 0-1.6 km (0-1 mi) and 0-80.5 km (0-50 mi) for either calculation. The cold release results and the hot release peak results are identical to those obtained using all 34 radionuclides in the full BR-2 inventory. The hot release mean values obtained with the 13 radionuclides removed differ by no more than 5 percent from the results obtained using 34 radionuclides. Thus, the 13 radionuclides for which acute dose conversion factors were not available do not contribute significantly to chronic dose or health effects, which suggests that none should have a significant impact on acute health effects.

The relative contributions to radiation exposures of the nuclides in an inventory can be estimated by normalizing the ratio of each nuclide's curie amount and the run 2a value by the sum of those ratios. A run 2a value is the curie amount of the radionuclide that produces significant radiation doses (IAEA, 1961; IAEA, 1990). The RADSEL code was used to perform this calculation for the set of 34 nuclides in the inventories used in this study. The RADSEL calculation showed that only one radionuclide, promethium-147, in the set of 13 nuclides for which acute health effect dose conversion factors were lacking, contributes significantly to dose at the 99.9 percent level. More importantly, the calculation also showed that promethium-147 accounts for only 0.5 percent of the total dose produced by the full set of 34 radionuclides. Thus, the 21 nuclides in the inventories for which acute health effect conversion factors were available account for all significant contributions to dose. Therefore, not including acute health effects for 13 of the 34 radionuclides in the inventories used in this study is not believed to have had a significant impact on the estimation of acute health effects, especially since none of these nuclides contributes significantly to chronic dose or health effects and since no acute effects were observed at any level including peak results for any calculation performed during this study.

D.5.4.3.3.3 Modal Study Cask Response Regions Release Fractions

The Modal Study (Fischer et al., 1987) developed release fractions for truck and rail accidents involving transportation cask containing commercial spent nuclear fuel. DOE as part of the preparation of the Programmatic SNF&INEL EIS, developed representative release fractions for metallic (aluminum-based) and nonmetallic (TRIGA) fuel for each of the Modal Study's cask response regions (DOE, 1995). Although there is not a direct relationship between the accident classification used in this EIS for ship accidents and that developed in the Modal Study, attempts were made to establish a meaningful comparison based on the definition of accidents and their consequences. Based on the accident definitions, one can approximate the severity category 5 ship accidents to the Modal Study's cask response region resulting from a medium impact mechanical force with a medium intensity thermal load. Table D-42 provides the values of release fractions used in this EIS for severity category 5 accident and that used for metallic and nonmetallic fuel in the Programmatic SNF&INEL EIS for a similar accident category. For ease of comparison, the EIS release fractions that were used in all of the base case calculations performed for this study are repeated in this table.

Table D-42 Programmatic SNF&INEL EIS Release Fractions

Element Group	Release Fraction		
		Programmatic SNF&INEL EIS	
	EIS (Base Case Category 5)	Metallic	Nonmetallic
Krypton	0.1	0.39	0.39
Cesium	9.0×10^{-4}	1.0×10^{-6}	0.00020
Ruthenium	1.0×10^{-6}	2.4×10^{-7}	0.000048
Particulate	5.0×10^{-8}	1.0×10^{-8}	0.0000020

Source: DOE, 1995

Inspection of the table shows that, except for the krypton element group, the base case EIS release fraction values for severity category 5 are somewhat larger than the values for nonmetallic fuel and are quite a bit larger than the values for metallic fuel. Thus, as would be expected, Table D-41 shows that mean and peak population doses and cancer fatalities for the distance ranges 0-1.6 and 0-80.5 km (0-1 and 0-50 mi) obtained using EIS release fractions are about five times larger than those obtained using nonmetallic fuel release fractions, which in turn are about 200 times larger than those obtained using metallic fuel release fractions. Therefore, since severity category 5 largely determines risk, use of EIS release fractions is conservative even if metallic and nonmetallic release fractions better represent releases during ship collisions.

D.5.4.3.3.4 Corrosion Products Release

During the operation of power reactors, radioactive cobalt is formed by neutron activation of chemical deposits on the outer surfaces of fuel rods. Thus, during transportation accidents, release of these radioactive deposits, usually referred to as corrosion products, can be a significant contributor to the size of the accident source term.

Because corrosion products formation is usually not a problem for research reactors, radioactive cobalt is not present in the inventories used in this study, and the sets of source terms input to MACCS do not contain fractions for corrosion products release. The potential impact of corrosion products release on foreign research reactor spent nuclear fuel accident source terms was examined by performing two sensitivity calculations. For these calculations, after scaling to match the size of the BR-2 inventory used in this study, the cobalt-60 content of the spent nuclear fuel inventory for a DOE test reactor (DOE, 1995) was added to the BR-2 inventory that was used in these sensitivity calculations (cobalt-58 was ignored as it should largely have decayed away before the fuel is shipped). Then, two sensitivity calculations were performed. Both calculations added 360 Ci of cobalt-60 to the BR-2 inventory and both used a value of 0.012 for the release fraction for the corrosion products chemical element group, as had been done in earlier studies. The first calculation examined the consequences of an accident that releases only corrosion products. Because corrosion products are not volatile, this release was assumed to be cold, that is driven by mechanical forces generated by the ship collision. The second calculation added the corrosion products release to the severity category 5 release used in the reference calculation. Because this release postulates a severe engulfing fire, the second calculation assumed that the release was hot.

Table D-41 shows that the first calculation, the cold release that contained only corrosion products (run 4a), leads to consequences that differ from those produced by the reference calculation as follows: for the 0-1.6 km (0-1 mi) distance range, mean values of population dose and cancer fatalities are about three times larger and peak values about two times smaller; for the 0-80.5 km (0-50 mi) distance range, mean and peak values for these two consequences are both smaller than the reference calculation results by factors of about 2.5 and 5 respectively. Mean and peak results for the 0-80.5 km (0-50 mi) distance range and peak results for the 0-1.6 km (0-1 mi) range are smaller because the curie content of the corrosion products release is smaller than the total curie content of the release used in the reference calculation (the release produced by severity category 5 release fractions and the BR-2 inventory). Mean results for the 0-1.6 km (0-1 mi) distance range are larger because the release is cold and therefore not lofted over nearby populations. Table D-41 also shows that adding the corrosion products release to the reference calculation (run 4b) increases consequence predictions only slightly (by about 20 percent), as would be expected given the small curie content of the corrosion products release compared to the reference release.

D.5.4.3.3.5 Work Force Population

Approximately 7,000 people work in Port Elizabeth in Newark, NJ. Thus, at least for accidents that occur during the workweek, these workers could be exposed to radiation as a result of a ship collision that involves a ship carrying foreign research reactor spent nuclear fuel. Inspection of maps showed that these workers should be added to the residential populations in the first distance intervals of the north sector of the Newark dock population distribution. Since the division of workers between these two distance intervals was not known, 3,500 workers were added to each interval for these sensitivity calculations.

Work force sensitivity calculations were performed first assuming, as was done for the reference calculation, a hot release, the BR-2 inventory, and severity category 5 release fractions. Then, this calculation was repeated two times assuming a cold release. The first of these two cold release calculations used the same shielding factors that had been used in the reference calculation. For the second cold release calculation, larger shielding factors were used during the first 24 hours after the accident over the distance range 0-8 km (0-5 mi) because the commercial buildings near the port are likely to provide better shielding than is provided by the mix of buildings located within 80.5 km (50 mi) of the port. Next, these two cold release calculations were repeated assuming that the release consists of a puff caused by the collision impact and a tail caused by the ensuing fire. Severity category 4 release fractions were used for the puff, and the release fractions for the tail were obtained by subtracting the severity category 4 release fractions from the severity category 5 release fractions. The puff was released when the collision occurred and lasted for 10 minutes; the tail was released one hour later and had a one hour release duration. Finally, the puff and tail calculation that did not use increased shielding factor values was repeated assuming that an evacuation would be called for should a severe accident lead to a fire that engulfed a radioactive material transportation cask, that the evacuation would begin about one hour after the accident took place (i.e., at about the time the tail release begins), and that the average evacuation speed would be slow because of city congestion.

Inspection of Table D-41 shows that, when a hot release is assumed (run 5a), adding a work force population increases mean population dose and cancer fatalities by less than a factor of 2 in the 0-1.6 km (0-1 mi) distance range, but has little effect on peak values in this distance range or on either mean or peak values in the 0-80.5 km (0-50 mi) distance range. When the release is cold (run 5b), 0-1.6 km (0-1 mi) mean population doses and cancer fatalities are increased by factors of about 26 and 2 respectively, and peak doses and cancer fatalities are increased by factors of about 3. For the 0-80.5 km (0-50 mi) distance range mean results are increased by factors of about 2 and peak results actually decrease by a factor of about 0.7. Moreover, these results are little changed by using increased shielding factors for commercial buildings, by assuming a puff and tail release, or by assuming a slow delayed evacuation.

The insensitivity to short-term shielding factor values, to release timing, and to evacuation is easy to understand when one remembers that population dose and cancer fatalities in these calculations are determined almost entirely by long-term groundshine exposures, which are of course little influenced by variation of any of these three short-term effects. Thus, as was shown above, elimination of lofting by assuming a cold release increases consequences, especially those that occur at short distances, but little else has much effect because only recovery actions (decontamination, temporary interdiction, condemnation) not examined by these sensitivity calculations can significantly affect long-term groundshine dose.

D.5.5 Port Accident Risk

The port accident risk analysis combines the results of the analysis of the frequency of ship accidents in the port area with the results of the consequence analysis of each of these accidents. Each of the accident severity categories contributes to the overall risk of accidents in the port. The total risk is the sum of the risk for each severity category. The specific methodology used to evaluate port accident risks and the results of that analysis are presented in this section.

The port accident risk analysis was performed based on 721 individual shipments of foreign research reactor spent nuclear fuel. Unlike the incident-free analyses, where the shipment of two or more casks on the same vessel results in an increase in the worker risk, the number of casks shipped on a single vessel does not affect the results of the analysis. The larger the number of casks on a single vessel, the fewer the number of shipments required to ship all 721 casks. Accident data is generated on a per transit basis. Assuming a single cask per shipment maximizes the number of shipments and maximizes the probability of an accident involving a ship carrying foreign research reactor spent nuclear fuel. If it is assumed that an accident that results in damage to a foreign research reactor spent nuclear fuel cask results in damage to all of the casks on a single vessel, the risks from the shipment of multiple casks on a single vessel would be identical to the risks associated with the shipment of the same number of casks individually. From the analysis performed in Appendix D Attachment D4, it is apparent that the probability of damage to all casks given that one is damaged in an accident is less than one. Therefore, performing the port accident risk analysis assuming that one cask is shipped per voyage results in an estimate of risk that is maximized for number of transportation casks shipped per voyage.

The accident risks have been evaluated for 13 ports: Elizabeth, NJ; the Hampton Roads, VA, ports of Portsmouth, Norfolk, and Newport News (using Portsmouth as the representative port); MOTSU, NC; Charleston, SC; Philadelphia. PA; Long Beach, CA; Savannah, GA; Galveston, TX; Concord NWS, CA; Tacoma, WA; Wilmington, NC; Jacksonville, FL; and Portland, OR. Although high population density ports do not meet the port selection screening criteria, the three high population ports of Elizabeth, Long Beach, and Philadelphia were included in the analysis for two purposes. First, it is possible that the shipments of foreign research reactor spent nuclear fuel could be made on vessels that make intermediate port calls, which could include these high population ports. Additionally, by evaluating these high population ports as ports of entry it was possible to estimate the maximum port accident risks resulting from the shipment of foreign research reactor spent nuclear fuel into the United States.

As discussed in the port accident consequence analysis (Section D.5.4), the accident analysis has evaluated the impact of accidents at two locations within each of the ports considered in the risk analysis. The two locations represent the possibility of: (1) an accident involving the ship transporting the foreign research reactor spent nuclear fuel while at the dock and (2) an accident at some point in the approach to the dock. Two locations were selected to address the possibility that the terminal (pier at which the cargo vessel is docked) may not be the location within the port that would yield the highest consequences for an accident. The key consideration is that in approaching the terminal, at some ports, the cargo vessel would pass through areas with a higher nearby population than the area around the terminal. To ensure that the accident consequence analysis did not underestimate the potential consequences, this second accident location was selected. It was selected by identifying the point in the approach to the terminal which had characteristics most likely to result in consequences representative of the largest consequences associated with an accident within the port facility. This generally meant a location near a population center. Accident locations were identified earlier in Table D-28.

Because two locations were selected for the accident analysis in each port, the total risk associated with a port call at the port of entry is the sum of the risks at these two locations. Accidents may occur either at the terminal (dock) or in the channel as the vessel approaches the dock This risk can be expressed as:

$$R_{PE} = \sum \left(M_D P_D + M_C P_C \right)$$

where:

R_{PE}=Risks from accidents in the port of entry,

M_D=Magnitude of the consequences for a severity category 4, 5, 6 accident at the dock,

P_D=Probability of an accident of severity category 4, 5, 6 at the dock,

M_C=Magnitude of the consequences for an accident in the approach to the dock (in the channel), and

P_C=Probability of an accident of severity category 4, 5, 6 in the approach to the dock (in the channel).

One of the assumptions made in the port risk analysis is that the vessel carrying the foreign research reactor spent nuclear fuel may make intermediate port calls at up to two different ports before arriving at the port of entry. In the event that these intermediate port calls are made, the risks associated with each of these port calls can be expressed as follows:

$$R_{IP} = \sum \left(M_D P_D + 2\ M_C P_C \right)$$

where R_{IP} is the risk from an accident in one of the intermediate ports of call. All other parameters have the same definitions as in the equation defining R_{PE}. The risks associated with accidents in the channel of the port is considered twice for the intermediate ports because the vessel must enter the harbor and approach the dock and, with the foreign research reactor spent nuclear fuel still on board, must depart the harbor. The accident frequency data is derived as a per transit frequency. For this risk analysis the approach to the dock has been considered to be part of one transit, the departure as part of a second transit.

From Section D.5.3.1.7, the probabilities per transit for the three accident severity categories evaluated are provided in Table D-43. These accident frequencies were used to develop the per transit probabilities for the accidents at the dock and in the channel for each of the intermediate ports and the ports of entry for the foreign research reactor spent nuclear fuel. The port accident data collected was not detailed enough to determine the percentage of accidents that occurred at the dock versus the percentage that occurred in the channel. For the purposes of this analysis, it was assumed that the accidents were evenly distributed between the dock and the approach to the dock. Table D-43 presents the per transit probabilities used in the port accident analysis for accidents at the dock and in the channel.

Table D-43 Port Accident Probabilities

Accident Severity Category	*P*	P_d	P_c
4	0.000006	0.000003	0.000003
5	0.000000005	0.0000000027	0.0000000027
6	0.0000000006	0.0000000003	0.0000000003

Accident consequences (mean results) for each of the accident severity categories are reproduced in Tables D-44 and D-45, in terms of total population dose and LCF, respectively. The consequences vary depending on the type of fuel involved in the accident, the port at which the accident occurs, the severity category, and the location of the accident within the port environs. The largest differences are between the different release categories and is the result of the smaller release fractions for a severity category 4 accident than for the severity category 5 and 6 accidents. Between the different ports assessed in the analysis, the consequences vary by a factor of approximately 30 [i.e., the consequences of an accident in Elizabeth (the location of the highest consequences) are approximately 30 times greater than the consequences of the same accident at MOTSU (the location with the lowest consequences)].

Using the equations presented previously in this section, the probability and consequence data were combined to generate the risk data presented in Table D-46. This table presents data on a per shipment basis and for the shipment of all 721 foreign research reactor spent nuclear fuel casks. Data is presented for shipments that are made with no intermediate port stops (identified as direct shipments in the table) and for shipments that are made with intermediate port stops. The direct shipments are quantified using the relationship developed for R_{PE}. For example, the risks in terms of person-rem associated with a single direct shipment consisting of a single cask of BR-2 fuel into the port of Elizabeth are the sum of the severity category 4 risks, severity category 5 risks, and severity category 6 risks associated with accidents at the Elizabeth dock (0.00000069, 0.000018, 0.0000020) and in the approach to the port of Elizabeth (0.0000011, 0.000019, 0.0000020), which is 0.000042 as shown in the table.

In developing the risk estimates for shipments that pass through intermediate ports, several combinations of intermediate ports were considered for each ultimate port of entry. The ports selected for use in this analysis represent the range of populations found in ports around the United States. As stated previously; Elizabeth, Philadelphia, and Long Beach are considered high population ports; Portland, Jacksonville, Tacoma, Concord NWS, and the Hampton Roads ports are considered to be intermediate population ports; and Charleston, Savannah, Wilmington, Galveston, and MOTSU are considered low population ports. Each possible combination of populations was considered for the intermediate ports. The risks associated

Table D-44 Port Accident Analysis—Total Effective Dose Equivalent Population Dose (Person-Rem)

Location	BR-2 Spent Nuclear Fuel Severity Category			RHF Spent Nuclear Fuel Severity Category			TRIGA Spent Nuclear Fuel Severity Category		
	4	5	6	4	5	6	4	5	6
Elizabeth (D)[1]	0.23	6600	6500	0.093	2600	2600	0.028	910	900
Elizabeth (C)[2]	0.38	6900	6800	0.15	2700	2700	0.045	960	940
Long Beach (D)[1]	0.21	4700	4800	0.085	1900	1900	0.025	650	660
Long Beach (C)[2]	0.081	4300	4400	0.032	1700	1700	0.0097	590	610
Philadelphia (D)[1]	0.18	2800	2800	0.071	1100	1100	0.021	380	380
Philadelphia (C)[2]	0.085	2700	2800	0.034	1100	1100	0.010	370	380
Portland (D)[1]	0.077	1200	1200	0.031	450	450	0.0093	160	160
Portland (C)[2]	0.053	1100	1200	0.021	430	440	0.0065	150	150
Norfolk (D)[1]	0.055	850	830	0.022	330	320	0.0067	110	110
Norfolk (C)[2]	0.030	670	660	0.012	250	250	0.0037	87	87
Charleston Wando Terminal (D)[1]	0.024	420	410	0.0096	150	150	0.003	53	53
Charleston NWS (D)[1]	0.016	480	480	0.0066	180	180	0.0021	61	61
Charleston (C)[2]	0.038	420	420	0.015	160	160	0.0046	54	54
Tacoma (D)[1]	0.056	1,700	1,800	0.022	670	700	0.0068	230	250
Tacoma (C)[2]	0.039	1,400	1,500	0.016	550	570	0.0048	190	200
Concord NWS (D)[1]	0.044	2,100	2,200	0.018	800	850	0.0054	280	300
Concord NWS (C)[2]	0.094	3,300	3,400	0.038	1,300	1,300	0.011	450	460
Jacksonville (D)[1]	0.028	680	680	0.011	260	250	0.0035	88	87
Jacksonville (C)[2]	0.026	530	550	0.010	200	200	0.0032	69	70
Savannah (D)[1]	0.056	490	500	0.022	180	180	0.0068	62	63
Savannah (C)[2]	0.013	380	390	0.005	140	140	0.0018	47	49
Wilmington (D)[1]	0.038	480	500	0.015	180	190	0.0047	62	64
Wilmington (C)[2]	0.0097	210	220	0.0038	75	80	0.0012	26	27
Galveston (D)[1]	0.073	1,400	1,600	0.029	550	600	0.0089	190	210
Galveston (C)[2]	0.032	1,400	1,600	0.013	540	590	0.0041	190	200
MOTSU (D)[1]	0.0073	210	220	0.0029	75	80	0.0010	25	27
MOTSU (C)[2]	0.0097	210	220	0.0038	75	80	0.0012	26	27

[1]*Accident is at the Dock*

[2]*Accident is in the Channel, the approach to the dock*

with a shipment that passed through two U.S. ports before arriving at the port of entry for the foreign research reactor spent nuclear fuel were calculated using the relationships for R_{PE} and R_{IP}. The risks were calculated for each intermediate port stop and added to the risks associated with operations within the port of entry, i.e., the risks associated with a direct shipment.

The per shipment data was used to calculate the risks associated with the basic implementation of Management Alternative 1 of proposed action. The values shown in the two rightmost columns of Table D-46 represent the risks associated with the shipment of all of the foreign research reactor spent nuclear fuel through a single port of entry via the same intermediate ports. Using the shipments through Elizabeth as an example, the value given for the program risks for the shipment of the foreign research reactor spent nuclear fuel through one intermediate and one low population port (0.027 person-rem or 0.000011 LCF) assumes that all 721 foreign research reactor spent nuclear fuel casks are shipped through these same three ports. The number of shipments of each type of fuel (473 BR-2, 86 RHF, and 162 TRIGA) were incorporated into the development of the risks.

Table D-45 Port Accident Analysis—Accident Consequences (LCF)

Location	BR-2 Spent Nuclear Fuel Severity Category 4	5	6	RHF Spent Nuclear Fuel Severity Category 4	5	6	TRIGA Spent Nuclear Fuel Severity Category 4	5	6
Elizabeth (D)[1]	0.00010	2.8	2.7	0.000041	1.1	1.1	0.000011	0.38	0.38
Elizabeth (C)[2]	0.00016	2.9	2.8	0.000066	1.1	1.1	0.000018	0.40	0.39
Long Beach (D)[1]	0.000093	2.0	2.0	0.000038	0.78	0.80	0.000010	0.27	0.28
Long Beach (C)[2]	0.000035	1.8	1.9	0.000014	0.71	0.73	0.0000040	0.25	0.26
Philadelphia (D)[1]	0.000078	1.2	1.2	0.000031	0.47	0.46	0.0000087	0.16	0.16
Philadelphia (C)[2]	0.000037	1.2	1.2	0.000015	0.45	0.47	0.0000042	0.16	0.16
Portland (D)[1]	0.000034	0.52	0.53	0.000014	0.20	0.20	0.0000039	0.068	0.069
Portland (C)[2]	0.000023	0.50	0.51	0.0000093	0.19	0.19	0.0000027	0.065	0.067
Norfolk (D)[1]	0.000024	0.38	0.37	0.0000097	0.14	0.14	0.0000028	0.049	0.048
Norfolk (C)[2]	0.000013	0.30	0.30	0.0000053	0.11	0.11	0.0000015	0.039	0.039
Charleston Wando Terminal (D)[1]	0.000011	0.19	0.19	0.0000042	0.070	0.070	0.0000012	0.024	0.024
Charleston NWS (D)[1]	0.0000068	0.22	0.22	0.0000027	0.080	0.080	0.00000084	0.028	0.028
Charleston(C)[2]	0.000017	0.19	0.19	0.0000067	0.070	0.071	0.0000019	0.024	0.024
Tacoma (D)[1]	0.000024	0.75	0.80	0.0000097	0.29	0.30	0.0000028	0.10	0.11
Tacoma (C)[2]	0.000017	0.63	0.66	0.0000068	0.24	0.25	0.0000020	0.083	0.087
Concord NWS (D)[1]	0.000019	0.90	0.96	0.0000076	0.34	0.37	0.0000022	0.12	0.13
Concord NWS (C)[2]	0.000041	1.4	1.5	0.000017	0.55	0.56	0.0000046	0.19	0.20
Jacksonville (D)[1]	0.000012	0.31	0.31	0.0000049	0.11	0.11	0.0000015	0.039	0.039
Jacksonville (C)[2]	0.000011	0.24	0.25	0.0000045	0.090	0.092	0.0000013	0.031	0.032
Savannah (D)[1]	0.000025	0.23	0.23	0.0000099	0.083	0.085	0.0000028	0.028	0.029
Savannah (C)[2]	0.0000059	0.18	0.19	0.0000023	0.065	0.067	0.00000074	0.022	0.023
Wilmington (D)[1]	0.000017	0.22	0.23	0.0000067	0.081	0.084	0.0000019	0.028	0.029
Wilmington (C)[2]	0.0000042	0.098	0.10	0.0000017	0.035	0.037	0.0000005	0.012	0.013
Galveston (D)[1]	0.000032	0.64	0.70	0.000013	0.24	0.27	0.0000037	0.084	0.092
Galveston (C)[2]	0.000014	0.63	0.69	0.0000056	0.24	0.26	0.0000017	0.082	0.090
MOTSU (D)[1]	0.0000032	0.099	0.11	0.0000013	0.035	0.038	0.00000041	0.012	0.013
MOTSU (C)[2]	0.0000042	0.098	0.10	0.0000017	0.035	0.037	0.00000052	0.012	0.013

[1] *Accident is at the Dock*

[2] *Accident is in the Channel, the approach to the dock*

Table D-46 Summary of Latent Cancer Fatalities and Population Exposure Risk—Per Shipment and for the Entire Program (Basic Implementation)

Port	Risks per Shipment: Population Exposure per Shipment (person-rem) BR-2	RHF	TRIGA	Risk per Shipment (LCF) BR-2	RHF	TRIGA	Program Risks: Expos. (person-rem)	Risk (LCF)
Elizabeth via:								
Two High Population Ports	0.00013	0.000052	0.000018	0.000000056	0.000000022	0.0000000075	0.070	0.000029
One High and One Intermediate Population Port	0.00011	0.000044	0.000016	0.000000048	0.000000019	0.0000000065	0.060	0.000025
One High and One Low Population Port	0.00011	0.000043	0.000015	0.000000045	0.000000018	0.0000000062	0.057	0.000024

Port	*Risks per Shipment*						*Program Risks*	
	Population Exposure per Shipment (person-rem)			*Risk per Shipment (LCF)*			*Expos. (person-rem)*	*Risk (LCF)*
	BR-2	*RHF*	*TRIGA*	*BR-2*	*RHF*	*TRIGA*		
Two Intermediate Population Ports	0.000056	0.000022	0.0000076	0.000000024	0.0000000093	0.0000000032	0.030	0.000013
One Intermediate and One Low Population Port	0.000051	0.000020	0.0000070	0.000000022	0.0000000085	0.0000000029	0.027	0.000011
Two Low Population Ports	0.000046	0.000018	0.0000063	0.000000020	0.0000000077	0.0000000026	0.024	0.000010
Direct	0.000042	0.000017	0.0000058	0.000000018	0.0000000070	0.0000000024	0.022	0.0000094
Long Beach via:								
Two High Population Ports	0.000011	0.000044	0.000015	0.000000047	0.000000018	0.0000000064	0.058	0.000025
One High and One Intermediate Population Port	0.000080	0.000032	0.0000011	0.000000034	0.000000013	0.0000000043	0.042	0.000018
One High and One Low Population Port	0.000071	0.000028	0.0000097	0.000000030	0.0000000012	0.0000000041	0.038	0.000016
Two Intermediate Population Ports	0.000050	0.000019	0.0000067	0.000000021	0.0000000083	0.0000000022	0.026	0.000011
One Intermediate and One Low Population Port	0.000041	0.000016	0.0000055	0.000000018	0.0000000068	0.0000000020	0.022	0.0000092
Two Low Population Ports	0.000032	0.0000013	0.0000043	0.000000014	0.0000000053	0.0000000018	0.017	0.0000072
Direct	0.000028	0.0000011	0.0000038	0.000000012	0.0000000046	0.0000000016	0.015	0.0000062
Philadelphia via:								
Two High Population Ports	0.00011	0.000042	0.000015	0.000000045	0.000000018	0.0000000061	0.057	0.000024
One High and One Intermediate Population Port	0.000088	0.000035	0.000012	0.000000037	0.000000015	0.0000000050	0.047	0.000020
One High and One Low Population Port	0.000083	0.000033	0.000011	0.000000035	0.000000014	0.0000000048	0.044	0.000019
Two Intermediate Population Ports	0.000031	0.000012	0.0000041	0.000000014	0.0000000052	0.0000000018	0.016	0.0000072

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One Intermediate and One Low Population Port	0.000026	0.000010	0.0000035	0.000000011	0.0000000044	0.0000000015	0.014	0.0000061
Two Low Population Ports	0.000021	0.0000083	0.0000028	0.0000000093	0.0000000036	0.0000000012	0.011	0.0000049
Direct	0.000017	0.0000069	0.0000023	0.0000000075	0.0000000029	0.0000000099	0.0092	0.0000040
Portland via:								
Two High Population Ports	0.000090	0.000035	0.000012	0.000000038	0.000000015	0.0000000050	0.047	0.000020
One High and One Intermediate Population Port	0.000059	0.000023	0.0000080	0.000000025	0.0000000099	0.0000000029	0.031	0.000013
One High and One Low Population Port	0.000050	0.000020	0.0000068	0.000000022	0.0000000084	0.0000000027	0.027	0.000011
Two Intermediate Population Ports	0.000029	0.000011	0.0000039	0.000000013	0.0000000049	0.00000000088	0.015	0.0000066
One Intermediate and One Low Population Port	0.000020	0.0000077	0.0000027	0.0000000090	0.0000000034	0.00000000068	0.011	0.0000048
Two Low Population Ports	0.000011	0.0000042	0.0000015	0.0000000051	0.0000000019	0.00000000049	0.0059	0.0000026
Direct	0.0000073	0.0000028	0.00000098	0.0000000032	0.0000000012	0.00000000026	0.0039	0.0000017
Norfolk via:								
Two High Population Ports	0.000095	0.000037	0.000013	0.000000040	0.000000016	0.0000000054	0.050	0.000021
One High and One Intermediate Population Port	0.000076	0.000030	0.000010	0.000000032	0.000000013	0.0000000043	0.040	0.000017
One High and One Low Population Port	0.000071	0.000028	0.0000097	0.000000030	0.000000012	0.0000000040	0.037	0.000016
Two Intermediate Population Ports	0.000019	0.0000071	0.0000024	0.0000000083	0.0000000031	0.0000000011	0.0098	0.0000044
One Intermediate and One Low Population Port	0.000014	0.0000052	0.0000018	0.0000000061	0.0000000023	0.00000000078	0.0072	0.0000032
Two Low Population Ports	0.0000088	0.0000033	0.0000011	0.0000000040	0.0000000015	0.00000000050	0.0046	0.0000021

Port	Risks per Shipment: Population Exposure per Shipment (person-rem)			Risks per Shipment: Risk per Shipment (LCF)			Program Risks	
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA	Expos. (person-rem)	Risk (LCF)
Direct	0.0000048	0.0000018	0.00000062	0.0000000021	0.0000000081	0.00000000028	0.0025	0.0000011
Charleston (Wando Terminal) via:								
Two High Population Ports	0.000092	0.000036	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.000010	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000016	0.0000063	0.0000021	0.0000000074	0.0000000028	0.00000000095	0.0087	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000043	0.0000015	0.0000000052	0.0000000019	0.00000000066	0.0061	0.0000027
Two Low Population Ports	0.0000066	0.0000024	0.00000082	0.0000000031	0.0000000011	0.00000000038	0.0035	0.0000016
Direct	0.0000027	0.000001	0.00000034	0.0000000012	0.00000000045	0.00000000015	0.0014	0.00000064
Charleston NWS via:								
Two High Population Ports	0.000093	0.000033	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.000010	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000017	0.0000063	0.0000022	0.0000000075	0.0000000028	0.00000000096	0.0084	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000044	0.0000015	0.0000000053	0.0000000020	0.00000000067	0.0058	0.0000028
Two Low Population Ports	0.0000068	0.0000025	0.00000084	0.0000000032	0.0000000011	0.00000000039	0.0032	0.0000017
Direct	0.0000028	0.0000011	0.00000036	0.0000000013	0.00000000048	0.00000000016	0.0011	0.00000068
MOTSU via:								
Two High Population Ports	0.000091	0.000036	0.000012	0.000000039	0.000000015	0.0000000052	0.048	0.000020

Port	Risks per Shipment: Population Exposure per Shipment (person-rem) BR-2	RHF	TRIGA	Risk per Shipment (LCF) BR-2	RHF	TRIGA	Program Risks: Expos. (person-rem)	Risk (LCF)
One High and One Intermediate Population Port	0.000072	0.000028	0.0000099	0.000000031	0.000000012	0.0000000041	0.038	0.000016
One High and One Low Population Port	0.000067	0.000026	0.0000092	0.0000000028	0.0000000011	0.0000000038	0.036	0.000015
Two Intermediate Population Ports	0.000015	0.0000057	0.0000019	0.0000000068	0.0000000025	0.00000000087	0.0080	0.0000036
One Intermediate and One Low Population Port	0.000010	0.0000038	0.0000013	0.0000000046	0.0000000017	0.00000000058	0.0054	0.0000024
Two Low Population Ports	0.0000053	0.0000019	0.00000064	0.0000000025	0.00000000088	0.0000000003	0.0028	0.0000013
Direct	0.0000013	0.00000047	0.00000016	0.00000000062	0.00000000022	0.000000000075	0.00069	0.00000032
Galveston via:								
Two High Population Ports	0.000099	0.000039	0.000013	0.000000042	0.000000016	0.0000000056	0.052	0.000022
One High and One Intermediate Population Port	0.000080	0.000031	0.000011	0.000000034	0.000000013	0.0000000046	0.042	0.000018
One High and One Low Population Port	0.000075	0.000029	0.0000010	0.000000032	0.0000000012	0.0000000043	0.040	0.000017
Two Intermediate Population Ports	0.000023	0.0000087	0.0000030	0.000000010	0.0000000038	0.0000000013	0.012	0.0000053
One Intermediate and One Low Population Port	0.000018	0.0000067	0.0000023	0.0000000080	0.000000003	0.000000001	0.0094	0.0000042
Two Low Population Ports	0.000013	0.0000048	0.0000017	0.0000000058	0.0000000022	0.00000000074	0.0068	0.0000031
Direct	0.0000090	0.0000034	0.0000012	0.0000000040	0.0000000015	0.00000000052	0.0047	0.0000021
Jacksonville via:								
Two High Population Ports	0.000094	0.000037	0.000013	0.000000040	0.000000016	0.0000000053	0.050	0.000021
One High and One Intermediate Population Port	0.000075	0.000029	0.00001	0.000000032	0.000000012	0.0000000043	0.040	0.000017

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
One High and One Low Population Port	0.000070	0.000027	0.0000096	0.000000029	0.000000012	0.0000000040	0.037	0.000016
Two Intermediate Population Ports	0.000018	0.0000067	0.0000023	0.0000000079	0.0000000030	0.0000000010	0.0093	0.0000041
One Intermediate and One Low Population Port	0.000013	0.0000048	0.0000016	0.0000000057	0.0000000021	0.00000000073	0.0067	0.000003
Two Low Population Ports	0.0000078	0.0000028	0.00000097	0.0000000036	0.0000000013	0.00000000045	0.0041	0.0000019
Direct	0.0000038	0.0000014	0.00000049	0.0000000017	0.00000000064	0.00000000022	0.0020	0.00000090
Savannah via:								
Two High Population Ports	0.000093	0.000037	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000074	0.000029	0.00001	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000069	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000017	0.0000063	0.0000021	0.0000000075	0.0000000028	0.00000000095	0.0088	0.0000039
One Intermediate and One Low Population Port	0.000012	0.0000044	0.0000015	0.0000000053	0.000000002	0.00000000067	0.0062	0.0000028
Two Low Population Ports	0.0000068	0.0000025	0.00000083	0.0000000032	0.0000000011	0.00000000039	0.0036	0.0000017
Direct	0.0000028	0.000001	0.00000035	0.0000000013	0.00000000048	0.00000000016	0.0015	0.00000069
Wilmington via:								
Two High Population Ports	0.000092	0.000036	0.000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000073	0.000029	0.00001	0.000000031	0.000000012	0.0000000042	0.039	0.000016
One High and One Low Population Port	0.000068	0.000027	0.0000094	0.000000029	0.000000011	0.0000000039	0.036	0.000015
Two Intermediate Population Ports	0.000016	0.0000061	0.0000021	0.0000000072	0.0000000027	0.00000000092	0.0084	0.0000038

Port	Risks per Shipment: Population Exposure per Shipment (person-rem) BR-2	RHF	TRIGA	Risk per Shipment (LCF) BR-2	RHF	TRIGA	Program Risks: Expos. (person-rem)	Risk (LCF)
One Intermediate and One Low Population Port	0.000011	0.0000042	0.0000014	0.0000000050	0.0000000019	0.00000000064	0.0058	0.0000026
Two Low Population Ports	0.0000062	0.0000022	0.00000076	0.0000000029	0.0000000010	0.00000000035	0.0032	0.0000015
Direct	0.0000022	0.00000082	0.00000028	0.0000000010	0.00000000037	0.00000000013	0.0012	0.00000053
Tacoma via:								
Two High Population Ports	0.000092	0.000036	0.0000013	0.000000039	0.000000015	0.0000000053	0.049	0.000021
One High and One Intermediate Population Port	0.000062	0.000024	0.0000084	0.000000026	0.000000010	0.0000000032	0.033	0.000014
One High and One Low Population Port	0.000053	0.000021	0.0000072	0.000000023	0.0000000088	0.0000000031	0.028	0.000012
Two Intermediate Population Ports	0.000031	0.000012	0.0000042	0.000000014	0.0000000053	0.0000000012	0.017	0.0000072
One Intermediate and One Low Population Port	0.000023	0.0000086	0.0000030	0.000000010	0.0000000038	0.00000000099	0.012	0.0000053
Two Low Population Ports	0.000014	0.0000052	0.0000018	0.0000000061	0.0000000023	0.00000000079	0.0072	0.0000032
Direct	0.0000097	0.0000038	0.0000013	0.0000000043	0.0000000016	0.00000000057	0.0051	0.0000023
Concord NWS via:								
Two High Population Ports	0.000099	0.000039	0.000013	0.000000042	0.000000016	0.0000000057	0.052	0.000022
One High and One Intermediate Population Port	0.000069	0.000027	0.0000093	0.000000029	0.000000011	0.0000000036	0.036	0.000015
One High and One Low Population Port	0.000060	0.000024	0.0000081	0.000000025	0.0000000099	0.0000000034	0.032	0.000013
Two Intermediate Population Ports	0.000038	0.000015	0.0000051	0.000000017	0.0000000064	0.0000000016	0.020	0.0000087
One Intermediate and One Low Population Port	0.000029	0.000011	0.0000039	0.000000013	0.0000000049	0.0000000014	0.016	0.0000067

Port	Risks per Shipment						Program Risks	
	Population Exposure per Shipment (person-rem)			Risk per Shipment (LCF)			Expos. (person-rem)	Risk (LCF)
	BR-2	RHF	TRIGA	BR-2	RHF	TRIGA		
Two Low Population Ports	0.000021	0.0000079	0.0000027	0.0000000090	0.0000000034	0.0000000012	0.011	0.0000047
Direct	0.000017	0.0000065	0.0000022	0.0000000071	0.0000000028	0.00000000096	0.0088	0.0000038

These risk estimates provide an estimate of the range of the port accident risks that would result from the basic implementation of Management Alternative 1 via the use of a wide range of ports. The ports of Elizabeth, Philadelphia, and Long Beach were included in the analysis as ports of entry even though they did not survive the port screening criteria. However, because of the high populations around these ports, their use provides an estimate of the highest risks associated with the shipment of foreign research reactor spent nuclear fuel into the United States. These risks can be contrasted with the risks associated with the shipment of the foreign research reactor spent nuclear fuel through MOTSU, which has an extremely low population around the port.

The port accident risks associated with the entire program range from a high of 0.070 person-rem and 0.000029 LCF, which assumes that all shipments would be made through two high population intermediate ports into Elizabeth, to a low of 0.0007 person-rem and 0.00000032 LCF, which assumes that the shipments are made directly into MOTSU. In the worst case analyzed the mean risks associated with port accidents results in an approximately one-in-a-thousand chance of a single LCF. The highest risks associated with a port that did meet the port selection criteria (assuming no restrictions on the selection of intermediate ports) is 0.000022 LCF. If, in addition, all intermediate port calls are restricted to port cities of similar size to those that meet the selection criteria, the highest calculated risk is reduced to 0.000009 LCF, approximately a one-in-a-hundred thousand chance of a single LCF.

D.5.6 Port Accident Impacts for Implementation Alternatives

Two implementation alternatives to Management Alternative 1 were identified that could impact the results of the port accident risk analysis that was developed for the basic implementation case. They are: 1a, Accepting Fuel from Developing Countries Only, and 2a, Accepting Fuel for Only Five Years. Developing countries are countries other than high income economies. Both of the implementation alternatives change the number of transportation casks containing foreign research reactor spent nuclear fuel that would be shipped to the United States.

The difference in the number of shipments does not affect the per-transit probability of an accident. The conditional probabilities of a severity category 4, 5, or 6 accident also do not change. On a per-shipment basis, the probability an accident of each of these severity categories is identical to the estimates used in the analysis of the basic implementation.

The consequences associated with each of the three accident severity categories also do not change just due to the change in the number of shipments. Since neither the probability nor the consequences of the accidents change, the per-shipment risks are identical to those of the basic implementation.

These alternatives are discussed in the following paragraphs.

Acceptance of Foreign Research Reactor Spent Nuclear Fuel from Developing Countries Only: Developing countries are defined as countries other than high-income economies. Under this alternative 168 transportation casks of foreign research reactor spent nuclear fuel would be shipped to the United States (see Appendix C.4.2 for details). All of these shipments would be shipped by ocean vessel and, therefore, would enter the United States through ports.

In addition to a reduced number of shipments associated with this alternative, the mix of fuel types changes. In the basic implementation of Management Alternative 1, most of the foreign research reactor spent nuclear fuel shipments would be BR-2 type fuel. Only about 20 percent of the shipments would be of the TRIGA fuel type. From the information provided in Appendix B, most of the shipments from countries other than high-income economies would be TRIGA fuel. Of the 168 shipments under this implementation alternative, 109 are TRIGA shipments. The remaining 59 shipments are BR-2 fuel shipments.

The risks of the basic implementation of Management Alternative 1, provided in Table D-46, have been recalculated to incorporate the change in the number and makeup of the shipments associated with this implementation alternative. These results are presented in Table D-47. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel through the port of Elizabeth via two high population intermediate ports. The port accident risks for this implementation alternative for this route are 0.0098 person-rem and 0.000004 LCF. The lowest calculated impacts are for the shipment of all of the material directly into MOTSU (no intermediate port calls) which results in port accident risks of 0.000095 person-rem and 0.000000045 LCF.

Table D-47 Summary of Risk and Population Exposure—For the Implementation Alternative of Acceptance of Foreign Research Reactor Spent Nuclear Fuel Only From Countries Other than High-Income Economies

Port	Exposure (person-rem)	Risk (LCF)
Elizabeth via:		
Two High Population Ports	0.0098	0.0000041
One High and One Intermediate Population Port	0.0084	0.0000035
One High and One Low Population Port	0.0080	0.0000034
Two Intermediate Population Ports	0.0041	0.0000018
One Intermediate and One Low Population Port	0.0038	0.0000016
Two Low Population Ports	0.0034	0.0000014
Direct	0.0031	0.0000013
Long Beach via:		
Two High Population Ports	0.0081	0.0000034
One High and One Intermediate Population Port	0.0059	0.0000025
One High and One Low Population Port	0.0052	0.0000022
Two Intermediate Population Ports	0.0037	0.0000015
One Intermediate and One Low Population Port	0.0030	0.0000013
Two Low Population Ports	0.0023	0.0000010
Direct	0.0021	0.00000087
Philadelphia via:		
Two High Population Ports	0.0079	0.0000033
One High and One Intermediate Population Port	0.0065	0.0000028
One High and One Low Population Port	0.0062	0.0000026
Two Intermediate Population Ports	0.0023	0.00000010
One Intermediate and One Low Population Port	0.0019	0.00000084

Port	Exposure (person-rem)	Risk (LCF)
Two Low Population Ports	0.0016	0.00000068
Direct	0.0013	0.00000055
Portland via:		
Two High Population Ports	0.0066	0.0000028
One High and One Intermediate Population Port	0.0044	0.0000018
One High and One Low Population Port	0.0037	0.0000016
Two Intermediate Population Ports	0.0021	0.00000085
One Intermediate and One Low Population Port	0.0015	0.00000060
Two Low Population Ports	0.00082	0.00000035
Direct	0.00054	0.00000022
Norfolk via:		
Two High Population Ports	0.0070	0.0000030
One High and One Intermediate Population Port	0.0056	0.0000024
One High and One Low Population Port	0.0052	0.0000022
Two Intermediate Population Ports	0.0014	0.00000061
One Intermediate and One Low Population Port	0.0010	0.00000045
Two Low Population Ports	0.00064	0.00000029
Direct	0.00035	0.00000016
Charleston (Wando Terminal) via:		
Two High Population Ports	0.0069	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00084	0.00000038
Two Low Population Ports	0.00048	0.00000022
Direct	0.00020	0.000000089
Charleston NWS via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00084	0.00000038
Two Low Population Ports	0.00048	0.00000022
Direct	0.00020	0.000000089
MOTSU via:		
Two High Population Ports	0.0067	0.0000028
One High and One Intermediate Population Port	0.0053	0.0000022
One High and One Low Population Port	0.0050	0.0000021
Two Intermediate Population Ports	0.0011	0.00000049
One Intermediate and One Low Population Port	0.00074	0.00000034
Two Low Population Ports	0.00038	0.00000018
Direct	0.000095	0.000000045
Galveston via:		
Two High Population Ports	0.0073	0.0000031
One High and One Intermediate Population Port	0.0059	0.0000025
One High and One Low Population Port	0.0055	0.0000023
Two Intermediate Population Ports	0.0017	0.00000074
One Intermediate and One Low Population Port	0.0013	0.00000058
Two Low Population Ports	0.00094	0.00000043
Direct	0.00066	0.00000029
Jacksonville via:		
Two High Population Ports	0.0069	0.0000029

Port	Exposure (person-rem)	Risk (LCF)
One High and One Intermediate Population Port	0.0055	0.0000023
One High and One Low Population Port	0.0052	0.00000022
Two Intermediate Population Ports	0.0013	0.00000057
One Intermediate and One Low Population Port	0.00093	0.00000042
Two Low Population Ports	0.00056	0.00000026
Direct	0.00028	0.00000013
Savannah via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0055	0.0000023
One High and One Low Population Port	0.0051	0.0000021
Two Intermediate Population Ports	0.0012	0.00000054
One Intermediate and One Low Population Port	0.00085	0.00000039
Two Low Population Ports	0.00049	0.00000023
Direct	0.00021	0.000000095
Wilmington via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0054	0.0000023
One High and One Low Population Port	0.0050	0.0000021
Two Intermediate Population Ports	0.0012	0.00000052
One Intermediate and One Low Population Port	0.00081	0.00000037
Two Low Population Ports	0.00045	0.00000021
Direct	0.00016	0.000000074
Tacoma via:		
Two High Population Ports	0.0068	0.0000029
One High and One Intermediate Population Port	0.0045	0.0000019
One High and One Low Population Port	0.0039	0.0000017
Two Intermediate Population Ports	0.0023	0.00000095
One Intermediate and One Low Population Port	0.0017	0.00000070
Two Low Population Ports	0.00010	0.00000045
Direct	0.00072	0.00000032
Concord NWS via:		
Two High Population Ports	0.0073	0.0000031
One High and One Intermediate Population Port	0.0051	0.0000021
One High and One Low Population Port	0.0044	0.0000019
Two Intermediate Population Ports	0.0028	0.0000012
One Intermediate and One Low Population Port	0.0022	0.00000091
Two Low Population Ports	0.0015	0.00000066
Direct	0.0012	0.00000053

Acceptance of Foreign Research Reactor Spent Nuclear Fuel for 5 Years Only: Under this implementation alternative, 586 transportation casks of foreign research reactor spent nuclear fuel would be shipped to the United States. All of these shipments would be shipped by ocean vessel and would enter the United States through ports.

In addition to a reduced number of shipments associated with this implementation alternative, the mix of fuel types changes slightly. From the information provided in Appendix B, 376 of the 586 shipments in this alternative are BR-2 spent fuel shipments, 56 are RHF, and 154 are TRIGA.

The risks of the basic implementation of Management Alternative 1, provided in Table D-46, have been recalculated to incorporate the change in the number and makeup of the shipments associated with this implementation alternative. These results are presented in Table D-48. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel through the port of Elizabeth via two high population intermediate ports. The port accident risks for the implementation alternative for this route are 0.055 person-rem and 0.000023 LCF. The lowest calculated impacts are for the shipment of all of the material directly into MOTSU (no intermediate port calls) which results in port accident risks of 0.00055 person-rem and 0.00000026 LCF.

Table D-48 Summary of Risk and Population Exposure—For the Implementation Alternative of a 5-Year Acceptance Duration

Port	*Exposure (person-rem)*	*Risk (LCF)*
Elizabeth via:		
Two High Population Ports	0.055	0.000023
One High and One Intermediate Population Port	0.047	0.000020
One High and One Low Population Port	0.045	0.000019
Two Intermediate Population Ports	0.023	0.000010
One Intermediate and One Low Population Port	0.021	0.0000091
Two Low Population Ports	0.019	0.0000082
Direct	0.018	0.0000074
Long Beach via:		
Two High Population Ports	0.046	0.000019
One High and One Intermediate Population Port	0.033	0.000014
One High and One Low Population Port	0.030	0.000013
Two Intermediate Population Ports	0.021	0.0000089
One Intermediate and One Low Population Port	0.017	0.0000073
Two Low Population Ports	0.013	0.0000057
Direct	0.012	0.0000049
Philadelphia via:		
Two High Population Ports	0.045	0.000019
One High and One Intermediate Population Port	0.037	0.000016
One High and One Low Population Port	0.035	0.000015
Two Intermediate Population Ports	0.013	0.0000057
One Intermediate and One Low Population Port	0.011	0.0000048
Two Low Population Ports	0.0089	0.0000039
Direct	0.0073	0.0000031
Portland via:		
Two High Population Ports	0.038	0.000016
One High and One Intermediate Population Port	0.025	0.000011
One High and One Low Population Port	0.021	0.0000090
Two Intermediate Population Ports	0.012	0.0000052
One Intermediate and One Low Population Port	0.0084	0.0000037
Two Low Population Ports	0.0047	0.0000021
Direct	0.0031	0.0000013
Norfolk via:		
Two High Population Ports	0.040	0.000017
One High and One Intermediate Population Port	0.032	0.000013
One High and One Low Population Port	0.030	0.000013
Two Intermediate Population Ports	0.0078	0.0000035
One Intermediate and One Low Population Port	0.0057	0.0000026
Two Low Population Ports	0.0036	0.0000017
Direct	0.0020	0.00000089
Charleston (Wando Terminal) via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0069	0.0000031

Port	Exposure (person-rem)	Risk (LCF)
One Intermediate and One Low Population Port	0.0048	0.0000022
Two Low Population Ports	0.0028	0.0000013
Direct	0.0011	0.00000051
Charleston NWS via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0066	0.0000031
One Intermediate and One Low Population Port	0.0048	0.0000022
Two Low Population Ports	0.0025	0.0000013
Direct	0.00087	0.00000054
MOTSU via:		
Two High Population Ports	0.038	0.000016
One High and One Intermediate Population Port	0.030	0.000013
One High and One Low Population Port	0.028	0.000012
Two Intermediate Population Ports	0.0063	0.0000028
One Intermediate and One Low Population Port	0.0042	0.0000019
Two Low Population Ports	0.0022	0.0000010
Direct	0.00055	0.00000028
Galveston via:		
Two High Population Ports	0.041	0.000018
One High and One Intermediate Population Port	0.033	0.000014
One High and One Low Population Port	0.031	0.000013
Two Intermediate Population Ports	0.0095	0.0000042
One Intermediate and One Low Population Port	0.0074	0.0000033
Two Low Population Ports	0.0054	0.0000024
Direct	0.0037	0.0000017
Jacksonville via:		
Two High Population Ports	0.039	0.000017
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0073	0.0000033
One Intermediate and One Low Population Port	0.0053	0.0000024
Two Low Population Ports	0.0032	0.0000015
Direct	0.0016	0.00000072
Savannah via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0069	0.0000031
One Intermediate and One Low Population Port	0.0049	0.0000022
Two Low Population Ports	0.0028	0.0000013
Direct	0.0012	0.00000055
Wilmington via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.031	0.000013
One High and One Low Population Port	0.029	0.000012
Two Intermediate Population Ports	0.0067	0.0000030
One Intermediate and One Low Population Port	0.0046	0.0000021
Two Low Population Ports	0.0026	0.0000012
Direct	0.00093	0.00000042

Port	Exposure (person-rem)	Risk (LCF)
Tacoma via:		
Two High Population Ports	0.039	0.000016
One High and One Intermediate Population Port	0.026	0.0000011
One High and One Low Population Port	0.022	0.0000094
Two Intermediate Population Ports	0.013	0.0000057
One Intermediate and One Low Population Port	0.0094	0.0000041
Two Low Population Ports	0.0057	0.0000026
Direct	0.0041	0.0000018
Concord NWS via:		
Two High Population Ports	0.041	0.000017
One High and One Intermediate Population Port	0.029	0.000012
One High and One Low Population Port	0.025	0.000011
Two Intermediate Population Ports	0.016	0.0000070
One Intermediate and One Low Population Port	0.012	0.0000054
Two Low Population Ports	0.0086	0.0000037
Direct	0.0070	0.0000030

D.5.7 Port Accident Impacts Associated with Management Alternative 2

Of the two subalternatives under Management Alternative 2, only subalternative 1b requires assessment of the impacts of accidents in port. This subalternative involves overseas reprocessing of foreign research reactor spent nuclear fuel. Under this subalternative, which is explained in detail in Chapter 2, up to eight transportation casks of vitrified high-level waste might pass through U.S. ports on their way to storage sites in the United States. The port accident impacts associated with this subalternative are evaluated below.

Foreign Reprocessing with Shipment of Vitrified Waste to a U.S. Storage Facility: In this subalternative to Management Alternative 2, all of the foreign research reactor spent nuclear fuel (including that generated in Canada) would be sent to either Great Britain or France for reprocessing and part or all of the vitrified high-level waste generated in the process could be shipped to the United States. Based on the reprocessing of approximately 23 metric tons of spent nuclear fuel (all of the fuel considered by the basic implementation of Management Alternative 1), enough vitrified high-level waste would be generated to require the transportation of up to eight transportation casks carrying logs of vitrified high-level waste to the United States.

The consequences of an accident in port involving a cask of vitrified high-level waste could not be derived from the analysis of the port accidents for the foreign research reactor spent nuclear fuel. Two significant differences in the contents of the cask carrying vitrified high-level waste and the casks carrying foreign research reactor spent nuclear fuel dictate that revised source terms be calculated for the vitrified high-level waste case. The release fractions associated with the accident severity categories are different for the vitrified high-level waste than they are for the foreign research reactor spent nuclear fuel. Based on previous DOE efforts (DOE, 1994b) the release fractions for vitrified high-level waste are the same for all three release categories (categories 4, 5, and 6). Vitrified waste release fractions are relatively insensitive to the affects of the fires that differentiate the category 5 and 6 accidents from the category 4 accidents. The release fractions used in this analysis are a factor of 0.05 higher than those used in the referenced analysis because the use of the MACCS code eliminates the need to describe a respirable fraction of the release. In the referenced analysis, the release fraction was determined and then modified by the respirable fraction (0.05) to use the value of 0.00000005 (5.0E-08) used in that analysis. Without the respirable fraction modification the release fraction is 0.000001 (1.0E-06). This is the release fraction used in the analysis of the vitrified high-level waste shipment port accident analysis.

These release fractions apply to all material in the vitrified high-level waste. Each isotope contained in the glassified waste has been assigned the same release fraction.

All of the wastes generated in reprocessing the foreign research reactor spent nuclear fuel would be transported in no more than eight casks, compared to the 837 marine and overland shipments of spent nuclear fuel required under the basic implementation of Management Alternative 1. This means that the curie content of the vitrified high-level waste could be approximately 100 times the content of a single transportation cask of foreign research reactor spent nuclear fuel.

In this analysis no credit has been taken for the reduction in the curie content of the vitrified high-level waste due to the natural decay that would result during the temporary storage of the vitrified high-level waste at the reprocessing facility. One of the options considered for this subalternative includes the storage of this material at the reprocessing facility until a permanent U.S. facility is ready to receive it for storage. Even if the material is not held until a permanent facility is available, some temporary storage at the reprocessing facility would probably be necessary. In either case, the reduction in the curie content of the waste logs has conservatively not been incorporated into this analysis. The risks associated with the shipment of aged vitrified high-level waste would be less, proportional to the reduction in the curie content, than the risks associated with the shipment of recently reprocessed material of the same volume. Therefore, while the risks calculated in this analysis are more appropriate for the shipment of recently reprocessed waste, the analysis bounds the risks associated with both options.

The isotopic content of the material shipped in one transportation cask of vitrified high-level waste is presented in Table D-49. This estimate was developed by combining the isotopic inventory of every assembly being shipped in the basic implementation of Management Alternative 1 and equally dividing these inventories into eight shipments. This inventory of material was developed from an earlier estimate of the number of foreign research reactor spent nuclear fuel shipments than that analyzed as the basic implementation of Management Alternative 1. The isotopic content of the earlier estimate of the number of shipments is slightly higher than results from the shipments in the basic implementation of Management Alternative 1, for every isotope found in the vitrified high-level waste. Therefore, the estimate used to generate the data in Table D-49 is slightly conservative.

The consequence analysis was performed using the MACCS code utilizing the inventory and release fraction data presented above. Since it is anticipated that the vitrified high-level waste would be stored, temporarily, at the Savannah River Site and the shipments are originating in Europe, only selected East Coast sites were analyzed. Port accident risks were analyzed for the ports of Philadelphia, Charleston, and MOTSU. Also, it has been assumed that the vitrified high-level waste shipments would be made on vessels that would not make intermediate port calls, i.e., on a chartered vessel. The results of these consequences analyses are presented in Table D-50. The highest mean value for an exposure to the MEI is 740 mrem for a 50-year dose to that individual. This corresponds to a LCF consequence of 0.00035.

The probability of an accident in port has been modeled using the data generated for the analysis of the basic implementation of Management Alternative 1. Although the use of a chartered (especially a purpose-built) ship could result in somewhat lower accident frequencies for each of the severity categories, these differences were judged to be minor and were not incorporated into the analysis. The port accident risks associated with the shipment of a single cask and of all eight casks containing the entire inventory of vitrified high-level waste generated in the reprocessing of all of the foreign research reactor spent nuclear fuel considered in the basic implementation of Management Alternative 1 are presented in Table D-51.

Table D-49 Radionuclide Inventory for Each of Eight Vitrified High-Level Waste Shipments

Radionuclide	*Vitrified High-Level Waste Inventory (Ci)*	*Radionuclide*	*Vitrified High-Level Waste Inventory (Ci)*
Hydrogen-3	7,302	Cerium-141	559,300
Krypton-85	207,000	Cerium-144 Promethium-144	24,890,000
Strontium-89	3,072,000	Promethium-147	3,703,000
Strontium-90 Yttrium-90	1,743,000	Promethium-148m	7,133
Yttrium-91	5,477,000	Europium-154	62,390
Zirconium-95	8,079,000	Europium-155	12,900
Niobium-95	16,540,000	Plutonium-238	8,484
Ruthenium-103 Rh-103m	716,000	Plutonium-239	405
Ruthenium-106 Rh-106m	1,882,000	Plutonium-240	326
Tin-123	33,340	Plutonium-241	78,440
Antimony-125	75,700	Americium-241	98
Tellurium-125m	18,060	Americium-242m	0.67
Tellurium-127M	69,720	Americium-243	1.4
Tellurium-129M	15,870	Curium-244	122
Cesium-134	1,413,000	Curium-242	990
Cesium-137	1,743,000		

Table D-50 Port Accident Consequences for Vitrified High-Level Waste

Location	*Mean Accident Consequences*		*99th Percentile Consequences*	
	Population Exposure (person-rem)	*LCF*	*Population Exposure (person-rem)*	*LCF*
MOTSU at the Dock	93.1	0.04	572	0.25
MOTSU in the Channel	66.1	0.029	332	0.13
Charleston at the Dock	202	0.088	747	0.32
Charleston in the Channel	293	0.13	2450	1.02
Philadelphia at the Dock	1250	0.54	5110	2.12
Philadelphia in the Channel	733	0.32	2990	1.21

The port accident risks associated with the implementation of this subalternative to Management Alternative 2 results in a negligible risk to the public. The highest mean port accident risk results in a less than one-in-ten thousand chance of a single LCF.

D.5.8 Port Accident Impacts Associated with a Combination of Returning Foreign Research Reactor Spent Nuclear Fuel and Overseas Management

In addition to evaluating the port accident impacts for the various alternatives associated with bringing all of the foreign research reactor spent nuclear fuel to the United States (Management Alternative 1) and managing all of the spent nuclear fuel overseas (Management Alternative 2), a hybrid scenario was analyzed. In this scenario, those countries that have the capability to store high-level waste would be encouraged to process aluminum-based foreign research reactor spent nuclear fuel and to accept the resulting high-level waste. For this scenario, those countries are assumed to be Belgium, France, Germany, Italy, Spain, Switzerland, and the United Kingdom. The United States would accept the foreign

Table D-51 Port Accident Risks for the Acceptance of Vitrified High-Level Waste

Port	Risk per Shipment of One Cask of Waste		Risk of the Entire Waste Acceptance Option	
	Population Dose (person-rem)	LCF	Population Dose (person-rem)	LCF
Philadelphia	0.006	0.000003	0.05	0.00002
Charleston	0.001	0.0000007	0.01	0.000005
MOTSU	0.0005	0.0000002	0.004	0.000002

research reactor spent nuclear fuel from those countries deemed not to have the high-level waste storage capability. In this option, this includes all of the countries identified in Table C-1, except for those listed above. Under the hybrid scenario, 452 shipments of spent nuclear fuel are assumed to be sent to the United States through U.S. ports, excluding shipments of Canadian origin, which are assumed to be transported overland. Of these, 290 are of the BR-2 fuel type and 162 are of the TRIGA type.

In analyzing the exposure and risk associated with this scenario, much of the information that was developed for Management Alternative 1 can be used. Both the per-transit probability of an accident and the conditional probabilities of severity category 4, 5, and 6 accidents are valid for this hybrid scenario. The consequences associated with each of the three accident severity categories also do not change, because the only thing that is changing is the number of shipments. Since neither the probability nor the consequences of the accidents change, the per-shipment risks are identical to those of the basic implementation of Management Alternative 1.

The risks associated with the basic implementation of Management Alternative 1 (Table D-46) have been recalculated to incorporate the change in the number and makeup of the shipments associated with the hybrid scenario. These results are presented in Table D-52. The highest calculated port accident risks are associated with the shipment of all of the foreign research reactor spent nuclear fuel through the port of Elizabeth via two high population intermediate ports. The port accident risks for the Management Alternative for this route are 0.041 person-rem and 0.000017 LCF. The lowest calculated impacts are for the shipment of all the material directly into MOTSU (no intermediate port calls), which results in port accident risk of 0.0004 person-rem and 1.9×10^{-7} LCF.

Table D-52 Summary of Risk and Population Exposure—For the Hybrid Scenario

Port	Exposure (person-rem)	Risk (LCF)
Elizabeth via:		
Two High Population Ports	0.041	1.7×10^{-5}
One High and One Intermediate Population Port	0.035	1.5×10^{-5}
One High and One Low Population Port	0.034	1.4×10^{-5}
Two Intermediate Population Ports	0.017	7.5×10^{-6}
One Intermediate and One Low Population Port	0.016	6.8×10^{-6}
Two Low Population Ports	0.014	6.1×10^{-6}
Direct	0.013	5.5×10^{-6}
Long Beach via:		
Two High Population Ports	0.034	1.5×10^{-5}
One High and One Intermediate Population Port	0.025	1.1×10^{-5}
One High and One Low Population Port	0.022	9.4 x 10-6
Two Intermediate Population Ports	0.015	6.6 x 10-6
One Intermediate and One Low Population Port	0.013	5.4 x 10-6
Two Low Population Ports	0.0099	4.3×10^{-6}
Direct	0.0087	3.7×10^{-6}
Philadelphia via:		
Two High Population Ports	0.033	1.4×10^{-5}

Port	Exposure (person-rem)	Risk (LCF)
Two High Population Ports	0.033	1.4×10^{-5}
One High and One Intermediate Population Port	0.028	1.2×10^{-5}
One High and One Low Population Port	0.026	1.1×10^{-5}
Two Intermediate Population Ports	0.0097	4.2×10^{-6}
One Intermediate and One Low Population Port	0.0082	3.6×10^{-6}
Two Low Population Ports	0.0066	2.9×10^{-6}
Direct	0.0054	2.3×10^{-6}
Portland via:		
Two High Population Ports	0.028	1.2×10^{-5}
One High and One Intermediate Population Port	0.018	7.8×10^{-6}
One High and One Low Population Port	0.016	6.7×10^{-6}
Two Intermediate Population Ports	0.0090	3.9×10^{-6}
One Intermediate and One Low Population Port	0.0063	2.7×10^{-6}
Two Low Population Ports	0.0035	1.6×10^{-6}
Direct	0.0023	9.8×10^{-7}
Norfolk via:		
Two High Population Ports	0.030	1.2×10^{-5}
One High and One Intermediate Population Port	0.024	1.0×10^{-5}
One High and One Low Population Port	0.022	9.3×10^{-6}
Two Intermediate Population Ports	0.0058	2.6×10^{-6}
One Intermediate and One Low Population Port	0.0043	1.9×10^{-6}
Two Low Population Ports	0.0027	1.2×10^{-6}
Direct	0.0015	6.7×10^{-7}
Charleston (Wando Terminal) via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.021	9.0×10^{-6}
Two Intermediate Population Ports	0.0051	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0036	1.6×10^{-6}
Two Low Population Ports	0.0021	9.5×10^{-7}
Direct	0.00083	3.8×10^{-7}
Charleston NWS via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.021	9.1×10^{-6}
Two Intermediate Population Ports	0.0049	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0034	1.6×10^{-6}
Two Low Population Ports	0.0019	9.8×10^{-7}
Direct	0.00041	4.0×10^{-7}
MOTSU via:		
Two High Population Ports	0.028	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.5×10^{-6}
One High and One Low Population Port	0.0021	8.8×10^{-6}
Two Intermediate Population Ports	0.0047	2.1×10^{-6}
One Intermediate and One Low Population Port	0.0032	1.4×10^{-6}
Two Low Population Ports	0.0016	7.6×10^{-7}
Direct	0.00041	1.9×10^{-7}
Galveston via:		
Two High Population Ports	0.031	1.3×10^{-5}
One High and One Intermediate Population Port	0.025	1.1×10^{-5}
One High and One Low Population Port	0.023	9.9×10^{-6}
Two Intermediate Population Ports	0.0071	3.2×10^{-6}

Port	Exposure (person-rem)	Risk (LCF)
One Intermediate and One Low Population Port	0.0056	2.5×10^{-6}
Two Low Population Ports	0.0040	1.8×10^{-6}
Direct	0.0028	1.2×10^{-6}
Jacksonville via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.9×10^{-6}
One High and One Low Population Port	0.022	9.2×10^{-6}
Two Intermediate Population Ports	0.0055	2.4×10^{-6}
One Intermediate and One Low Population Port	0.0039	1.8×10^{-6}
Two Low Population Ports	0.0024	1.1×10^{-6}
Direct	0.0012	5.3×10^{-7}
Savannah via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.7×10^{-6}
One High and One Low Population Port	0.022	9.1×10^{-6}
Two Intermediate Population Ports	0.0052	2.3×10^{-6}
One Intermediate and One Low Population Port	0.0036	1.7×10^{-6}
Two Low Population Ports	0.0021	9.8×10^{-7}
Direct	0.00088	4.1×10^{-7}
Wilmington via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.023	9.6×10^{-6}
One High and One Low Population Port	0.021	9.0×10^{-6}
Two Intermediate Population Ports	0.0050	2.2×10^{-6}
One Intermediate and One Low Population Port	0.0035	1.6×10^{-6}
Two Low Population Ports	0.0019	8.9×10^{-7}
Direct	0.00069	3.2×10^{-7}
Tacoma via:		
Two High Population Ports	0.029	1.2×10^{-5}
One High and One Intermediate Population Port	0.019	8.2×10^{-6}
One High and One Low Population Port	0.016	7.0×10^{-6}
Two Intermediate Population Ports	0.0098	4.2×10^{-6}
One Intermediate and One Low Population Port	0.0070	3.1×10^{-6}
Two Low Population Ports	0.0043	1.9×10^{-6}
Direct	0.0030	1.3×10^{-6}
Concord NWS via:		
Two High Population Ports	0.031	1.3×10^{-5}
One High and One Intermediate Population Port	0.021	9.1×10^{-6}
One High and One Low Population Port	0.019	7.9×10^{-6}
Two Intermediate Population Ports	0.012	5.1×10^{-6}
One Intermediate and One Low Population Port	0.0092	4.0×10^{-6}
Two Low Population Ports	0.0064	2.8×10^{-6}
Direct	0.0052	2.2×10^{-6}

D.5.9 Consequences of Sabotage or Terrorist Attack

This section provides an evaluation of impacts that could potentially result from a malicious act on a shipment of foreign research reactor spent nuclear fuel. In no instance, even in severe cases such as those discussed below, could a nuclear explosion or permanent contamination of the environment leading to condemnation of land occur. Furthermore, DOE considers that, due to the security measures that would be in place for any spent nuclear fuel shipments, such attacks would be unlikely to occur. At a minimum, the extent or effects of any such attacks, would be mitigated by the security measures.

Since it is impossible to determine with certainty the probability of a deliberate act of sabotage or terrorist attack, this section presents an analysis of potential consequences of sabotage or terrorist attack on a spent nuclear fuel shipping cask, and does not attempt to estimate the risk of such an activity. Although judged very unlikely to actually occur, a malicious attack on a foreign research reactor spent nuclear fuel shipping cask has been postulated to occur at a U.S. port or during transportation from the port to the management site, for purposes of illustrating the effects that might result from such an event.

The spectrum of attacks that can be postulated is broad, falling into three categories or scenarios: (1) exploding a bomb near a shipping cask, (2) attacking a cask with a shaped charge, or an armor-piercing weapon (i.e., an anti-tank weapon), and (3) hijacking (stealing) a shipping cask. None of the scenarios considered would lead to a criticality accident.

D.5.9.1 Exploding a Bomb Near a Shipping Cask

This sabotage/terrorist attack scenario assumes that a large bomb, similar to that detonated in Oklahoma City in April of 1995, is detonated in the immediate vicinity of a spent nuclear fuel shipping cask. The primary threats to the cask integrity would arise from: (1) direct blast forces (shock wave) from the bomb, (2) impact forces from fragments (e.g., motor vehicle parts) generated by the bomb, and (3) other dynamic forces such as a roll-over of the cask transport vehicle in response to the blast forces. The casks are rugged structures that would be expected to survive the effects of a nearby bomb explosion with no significant loss of integrity. At worst, the blast might produce a crack in the wall of the cask. In any case, all spent nuclear fuel elements would remain inside the cask. Blast-related damage might, however, reduce the effectiveness of cask shielding and/or cause locally higher dose rates outside the cask (e.g., from damaged shielding areas and radiation streaming through a crack in the cask wall).

Although no mechanism has been postulated that could cause such an event, an analysis of a total loss of cask shielding has been performed for the purposes of demonstrating limiting case effects of an attack on a spent nuclear fuel shipping cask, such as that discussed above. The analysis scenario assumes that the cask was full of a highly irradiated foreign research reactor spent nuclear fuel, and that the spent nuclear fuel elements were spread on the ground producing the highest possible direct dose rate. For the calculation of direct dose, no credit was taken for self-shielding of the spent fuel, and it was assumed that no other obstacle would exist between the spent nuclear fuel and individual members of the public. Since the spent nuclear fuel would be a solid metal structure, this analysis assumes that no spent nuclear fuel damage occurs, therefore, no radioactive materials would be dispersed. The results of this unrealistically conservative analysis are shown in Figure D-60. This figure provides a conservative estimate of the direct dose rate (rem per hour) to an individual member of the public versus distance from a spent nuclear fuel pile consisting of 30 highly irradiated fuel elements. Based on the results of this hypothetical, conservative analysis, an evacuation distance of about 900 meters (3000 ft) would be sufficient to maintain a dose rate of less than 10 mrem per hour, (or 0.01 rem per hour). This is a very conservative evacuation distance, but it would provide a good measure for consideration by an emergency response team. This scenario would result in minimal or no contamination of the area where it occurred and once the spent nuclear fuel was shielded, the evacuation zone would be greatly reduced. Once the spent nuclear fuel was removed from the site, the area would be decontaminated, if necessary, before it returned to normal.

D.5.9.2 Attacking a Cask with a Shaped Charge or Armor-Piercing Weapon

If a cask were attacked by an armor-piercing weapon or a shaped charge, the cask would be penetrated and spent nuclear fuel elements inside the cask could be damaged. An analysis of a hypothetical attack on a spent nuclear fuel shipping cask using a shaped charge was performed using the MACCS code. The

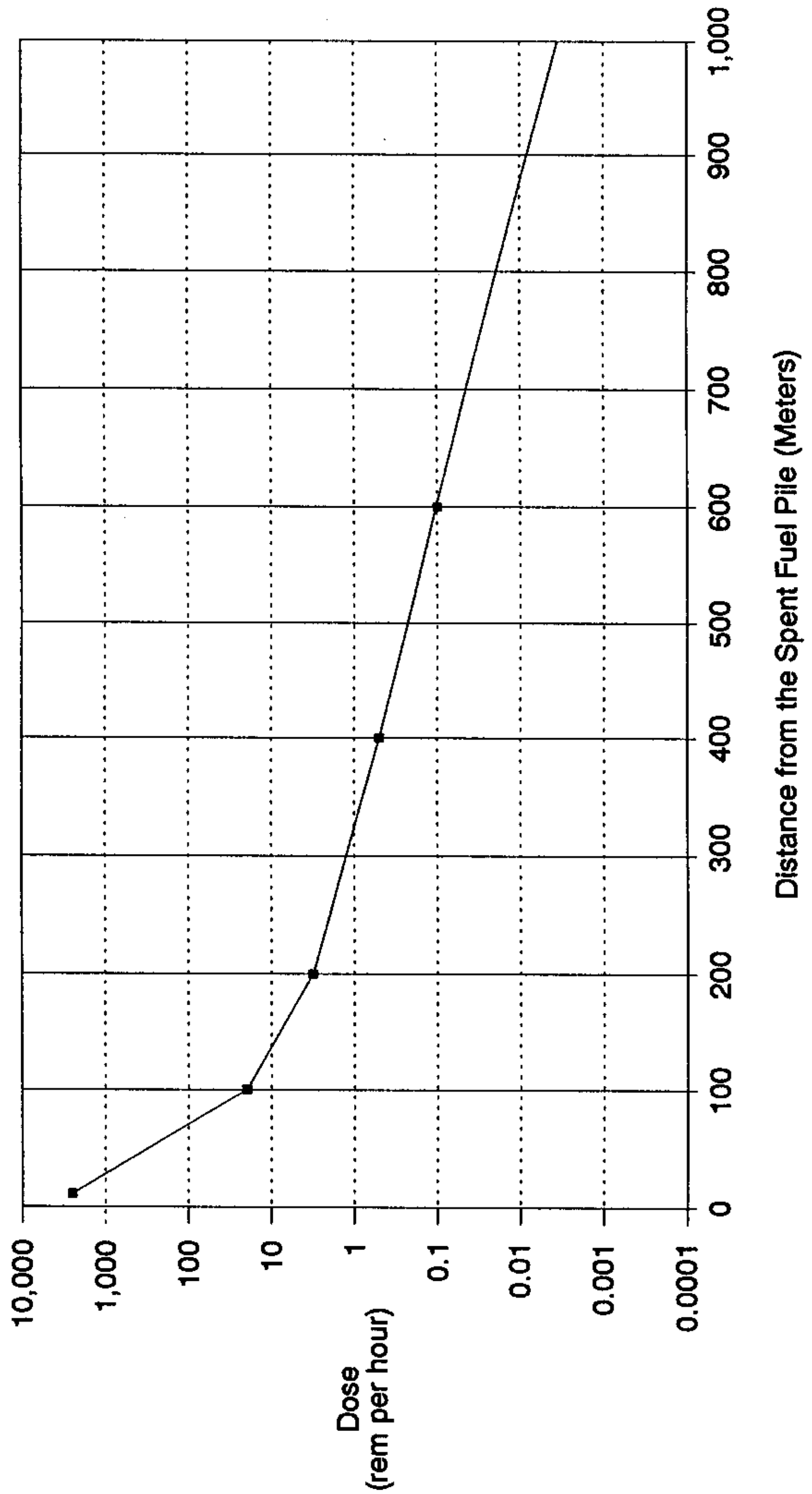


Figure D-60 Direct Dose vs Distance to an Individual Member of the Public

accident was assumed to occur on a city street in a highly populated area near the harbor where the spent nuclear fuel cask was transferred to a truck after trans oceanic shipment from overseas. The analysis assumed that the cask contained the highest radionuclide inventory, and the blast released all of the noble gases and one percent of the bulk of the spent nuclear fuel as airborne aerosols. The one percent of bulk spent nuclear fuel release assumption was based on measurements of aerosols released during tests where spent nuclear fuel was explosively disrupted. These tests yielded spent nuclear fuel release mass fractions that ranged from 0.05 to 2.5 percent (Sanders, et al., 1992). The blast energy would be quickly dissipated and the released fission products and gases and aerosols were assumed to be relatively cool; thus no plume rise was assumed to occur. These assumptions are very conservative and the results provide an enveloping estimate of consequences on the environmental and health effects. The MACCS calculations estimated a population dose of 208,000 person-rem with no acute fatalities or short-term adverse health effects among the exposed population. The MACCS results estimated that 91 latent cancer fatalities could occur among the 16 million persons living within 80 kilometers (50 miles) of the attack. The average individual lifetime radiation dose among the one to two million people who would be exposed is estimated to be about 200 mrem. This is less than one percent of a person's lifetime natural background radiation dose. This evaluation did not consider any evacuation and/or sheltering activities after the attack. MACCS also estimated a contamination distance of about 1 kilometer (0.6 miles) down wind from the attack. This distance, though conservative, could be used by an emergency response team for evacuation purposes. Of course, any actual evacuation distance would be determined on a case-by-case basis, if such an event were ever to occur. Mitigation activities in the aftermath of such an explosion, as required by law (EPA), would reduce the size of the contaminated area drastically and the area could become rehabitable in a short period of time. It is important to bear in mind that the explosion itself would be likely to produce fatalities, injuries and property damage that far exceed that caused by any release of radioactive material from the spent nuclear fuel.

In a terrorist attack using an anti-tank weapon, any cask damage and resulting consequences would be less severe than the accidents analyzed elsewhere in the EIS. This is because (1) there would be no explosive material inside the cask so the cask would not explode. Therefore, no additional radioactivity, other than that released directly by the projectile, would be forced out of the cask, and (2) there would be no fire to disperse the radioactivity that would be released when the cask was breached. At worst, the consequences of a terrorist attack on a spent nuclear fuel shipping cask with an anti-tank weapon would be similar to that analyzed above for a hypothetical terrorist attack on a cask with a high explosive shaped charge.

D.5.9.3 Hijacking a Shipping Cask

The discret theft of a spent nuclear fuel transportation cask is considered to be very unlikely, due to security measures that would be in place during transportation activities, especially the guarding of the cask, and communication and tracking systems (see Section 2.8 and Appendix H). In addition, the large size and weight of these casks (20 to 30 metric tons) and the inherent radioactivity of the spent nuclear fuel (which could kill a person upon contact) would deter most would-be hijackers. In the event of a hijack attempt, required communications systems would ensure timely notification of authorities who would mobilize response forces. The installed tracking system would allow the location of the cask to be determined in real time, thereby aiding timely interception of hijackers by response forces.

No release of radioactive material or increase in radiation level would be expected during a hijack scenario unless the hijacker could blow up the cask using explosive material (e.g., a shaped charge), or open the cask. In case of a cask explosion using a shaped charge, the consequences would be the same as, or smaller than (depending on the location of the accident), the case described in Section D.5.9.2. If the cask were opened (a lengthy process requiring special tooling), shielding would be decreased and the radiation level in the immediate vicinity of the cask would increase. The cask opening could only be accomplished at great personal risk to hijackers due to large (possibly immediately lethal) radiation exposures that they would receive while handling the unshielded fuel elements.

Should such an attempt be made, the hijackers would not be able to alter the fuel configuration inside the cask to make it critical. Criticality analyses that have been performed in support of the cask certification process consider various fuel and moderation configurations. These analyses are performed to ensure that none of potential configurations that could occur during loading and transport of the cask would lead to a criticality condition. Changing moderating material to achieve criticality, would require special materials that are not readily available (safeguard materials). Based on the time available to the hijackers, and tooling and materials that are needed, DOE considers that the potential for achieving criticality in a hijacked spent nuclear fuel cask is beyond credibility. If the hijackers were to dump the unshielded spent nuclear fuel, the resulting consequences to the public from the bare spent nuclear fuel radiation exposure would be less severe than those already analyzed for other hypothetical scenarios in this appendix.

References

AAPA (American Association of Port Authorities), 1994, *The 1994 AAPA Directory, Seaports of the Americas,* Compass North America, Inc., Coral Gables, FL.

AAPA (American Association of Port Authorities), 1993, *The 1993 AAPA Directory, Seaports of the Western Hemisphere,* Compass North America, Inc., Coral Gables, FL.

Abkowitz, M. and J. Galarraga, 1985, "Tanker Accident Rates and Expected Consequences in U.S. Ports and High Seas Regions", *Conference on Recent Advances in Hazardous Materials Transportation Research: An International Exchange,* Transportation Research Council, November 10-13.

Adams, D., 1994, Chief Wharfinger, Port of Oakland, CA, Hazardous Cargo DADAMS Data Radioactive - 1994, June 7.

Adams, D. and H. Renteria, 1994, Chief Wharfinger and Emergency Services Manager, Port of Oakland, CA, personal communication with R. L. Gotchy, Science Applications International Corporation, June 7.

Adams, D., 1993, Chief Wharfinger, Port of Oakland, CA, personal communication with C. Miller, Science Applications International Corporation, October 19.

American Shipper, 1994, *American Shipper, Southern Ports, 1994,* An American Shipper Publication, Volume 35, No. 13, Jacksonville, FL, January.

Babrauskas, V., 1986a, "Room Fire Temperature Computations," *Fire Protection Handbook,* 16th Edition, A.E. Cote and J. L. Linville, Editors, National Fire Protection Association, Quincy, MA.

Babrauskas, V., 1986b, "Pool Fires: Burning Rates and Heat Fluxes," *Fire Protection Handbook,* 16th Edition, A.E. Cote and J. L. Linville, Editors, National Fire Protection Association, Quincy, MA.

Banks, R., 1994, Field Supervisor, U.S. Fish and Wildlife Service, Letter to D. Scott, Science Applications International Corporation, October 26.

BEIR (Biological Effects of Ionizing Radiation), 1990, *Health Effects of Exposure to Low Levels of Ionizing Radiation,* BEIR V, National Academy Press, Washington, DC.

Benedict, M., T. Pigford and H. Levi, 1981, *Nuclear Chemical Engineering,* McGraw-Hill, New York.

Benham, K., W. Friedman and J. Kelly, 1994, Manager of Health and Safety, Manager of Marine Communications, and Manager of Marine Operations, Port of Seattle, WA, personal communication with R. L. Gotchy and D. Amick, Science Applications International Corporation, June 20.

Benham, K. and G. Schuler, 1993, Marine Operations and Safety Officer, Port of Seattle, WA, personal communication with C. Miller, Science Applications International Corporation, October 18.

Bennett, R. F., 1995, *Twenty-Five Questions (and Answers) About Marine Pilotage in South Carolina,* Charleston Branch Pilots' Association, Charleston, SC, April.

Bentzien, M., 1994, Assistant Field Supervisor, U.S. Fish and Wildlife Service, Jacksonville, FL, Letter to D. Scott, Science Applications International Corporation, November 15.

Betz, D., 1994, Mayor of Wilmington, NC, Comments on Draft EA of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel, personal communication to T. Grumbly, Assistant Secretary of Energy, U.S. Department of Energy, March 7.

Bolt, B., 1978, *Earthquakes - A Primer,* W. H. Freeman and Company, San Francisco, CA.

Boyer, M., 1994, Pennsylvania Department of Environmental Resources (PDER), Water Quality Section, personal communication to D. Scott, Science Applications International Corporation, August.

Breslin, S., 1993, Texas Natural Heritage Program, Texas Parks and Wildlife Department, Austin, TX, personal communication to M. Shafer, Science Applications International Corporation, September 15.

Brooks, K. E., 1994, Marketing Director, Port of Wilmington, DE, personal communication with D. Amick, Science Applications International Corporation, November 30.

Brown, M., 1995, U.S. Coast Guard, Mobile, AL, personal communication with L. Danese, Science Applications International Corporation, January 18.

Casper, E., 1993, Director of Operations, Port of Wilmington, DE, personal communication with C. Miller, Science Applications International Corporation, November 12.

Castagnola, A., 1994, Director of Marketing, South Jersey Port Corporation, Camden, NJ, personal communication with C. Miller, Science Applications International Corporation, August 8.

Castagnola, A., 1993, Director of Marketing, South Jersey Port Corporation, Camden, NJ, personal communication with C. Miller, Science Applications International Corporation, October 18.

Castiel, A., 1993, Marketing Services Coordinator, Jacksonville Port Authority, Jacksonville, FL, personal communication to C. Miller, Science Applications International Corporation, October 19.

Commissioners of the Pilotage, 1995, *Policy and Procedures Manual*, Commissioners of the Pilotage, Port of Charleston, Charleston, South Carolina, June.

Cordfunke, E. H. P. and R. J. M. Konings, 1990, *Thermochemical Data for Reactor Material and Fission Products*, North Holland, Amsterdam, Holland, p. 105.

Cubicoitti, D. and B. R. Sehgal, 1984, "Vapor Transport of Fission Products in Postulated Severe Light Water Reactor Accidents," *Nuclear Technology*, 65 (2).

Dalton, A., 1994, Delaware Natural Heritage Inventory, Delaware DNREC, Division of Fish and Wildlife, Dover, DE, personal communication to D. Scott, Science Applications International Corporation, September 20.

Davis, P., 1994, Manager, Penn Terminals, Eddystone, PA, personal communication with C. Miller, Science Applications International Corporation, August.

Department of the Navy, 1994, *Environmental Assessment, Construction of Magazines and Rail Lines, Naval Weapons Station, Charleston, South Carolina*, Public Works Officer, Naval Weapons Station Charleston, Charleston, SC, July.

Department of the Navy, 1990, *Environmental Assessment for Propulsion Training Facility, Naval Weapons Station Charleston, Berkeley County, South Carolina*, Southern Division, Naval Facility Engineering Command, Charleston, SC, July.

DOC (U.S. Department of Commerce), 1994, Bureau of Census, Statistical Abstract of the United States (114th edition), Washington, DC.

DOC (U.S. Department of Commerce), 1993a, *United States Coast Pilot; Atlantic Coast: Eastport to Cape Cod*, Volume 1, 28th ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOC (U.S. Department of Commerce), 1993b, *United States Coast Pilot; Atlantic Coast: Cape Cod to Sandy Hook*, Volume 2, 27th ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOC (U.S. Department of Commerce), 1993c, *United States Coast Pilot; Atlantic Coast: Sandy Hook to Cape Henry*, Volume 3, 30th ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOC (U.S. Department of Commerce), 1993d, *United States Coast Pilot; Atlantic Coast: Cape Henry to Key West,* Volume 4, 29th ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOC (U.S. Department of Commerce), 1992a, *United States Coast Pilot; Atlantic Coast: Gulf of Mexico, Puerto Rico, and Virgin Islands,* Volume 5, 23rd ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOC (U.S. Department of Commerce), 1992b, *United States Coast Pilot; Pacific Coast: California, Oregon, Washington, and Hawaii,* Volume 7, 27th ed., National Oceanic and Atmospheric Administration, Washington, DC.

DOE (U.S. Department of Energy), 1995, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement,* DOE/EIS-0203-F, Idaho Operations Office, April.

DOE (U.S. Department of Energy), 1994a, *Implementation Plan for the Environmental Impact Statement on a Proposed Policy for the Acceptance of Foreign Research Reactor Spent Nuclear Fuel,* DOE/EIS-0218, October.

DOE (U.S. Department of Energy), 1994b, *Savannah River Site, Defense Waste Processing Facility Safety Analysis Report—200-S Area,* DPSTSA-200-10, Supplement 20, June.

DOE (U.S. Department of Energy), 1994c, *Terms and Conditions for Financial Settlement for Receipt and Disposition of Foreign Research Reactor Spent Fuels,* Savannah River Operations Office, revised May 23.

DOE (U.S. Department of Energy), 1994d, *Environmental Assessment of Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel,* DOE/EA-0912, Washington, DC, Idaho Falls, Idaho, April.

DOE (U.S. Department of Energy), 1993, "Notice of Intent to Prepare an Environmental Impact Statement on a Proposed Policy for the Acceptance of United States Origin Foreign Research Reactor Spent Nuclear Fuel," *Federal Register,* p. 54336, October 21.

DOE (U.S. Department of Energy), 1988a, *External Dose-Rate Conversion Factors for Calculation of Dose to the Public,* DOE/EH-0070, Washington, DC, July.

DOE (U.S. Department of Energy), 1988b, *Internal Dose-Rate Conversion Factors for Calculation of Dose to the Public,* DOE/EH-0071, Washington, DC, July.

D&B (Dun & Bradstreet Information Services), 1993, *Exporters' Encyclopedia,* Port City Press, Baltimore, MD.

DRBC (Delaware River Basin Commission), 1994, *Delaware River and Bay Water Quality Assessment, 1992-1993 305(b) Report,* West Trenton, NJ, May.

Edwards, G. and J. Drews, 1993, Public Relations Officer, and Norfolk International Terminal Safety Officer, Port of Hampton Roads, Virginia Port Authority, VA, personal communication with C. Miller, Science Applications International Corporation, October 14.

Edwards, W. W. and R. A. Burns, 1993, Executive Director and Assistant Executive Director, Port of Gulfport, MS, personal communication with C. Miller, Science Applications International Corporation, November 29.

Ellison, P. G., 1993, "Aluminum-Uranium Fuel-Melt Behavior During Severe Nuclear Reactor Accidents," *Nuclear Safety,* Volume 34, No. 2, April-June.

EPA (U.S. Environmental Protection Agency), 1991, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,* EPA 400-R-92-001, Office of Radiation Programs, Washington, DC, October (Reprinted May 1992).

Fabens, D., 1994, Louisiana Department of Environmental Quality, personal communication with D. Scott, Science Applications International Corporation, November 11.

FHI (Frederick R. Harris, Inc.), 1994a, *Yorktown Naval Weapons Station, Yorktown, Virginia; Assessment for Receipt of Foreign Research Reactor Spent Nuclear Fuel,* Final Report, Iselin, NJ, January 17.

FHI (Frederick R. Harris, Inc.), 1994b, *Ports of Virginia Ports Authority; Norfolk Terminal, Portsmouth Terminal, Newport News Terminal, Assessment for Receipt of Foreign Research Reactor Spent Nuclear Fuel,* Iselin, NJ, January 17.

FHI (Frederick R. Harris, Inc.), 1993a, *Port of Charleston, Assessment for Receipt of Foreign Research Reactor Spent Nuclear Fuel,* Iselin, NJ, August 31.

FHI (Frederick R. Harris, Inc.), 1993b, *Kings Bay Naval Submarine Base; Assessment for Receipt of Foreign Research Reactor Spent Nuclear Fuel,* Final Report, Iselin, NJ, November 1.

FHI (Frederick R. Harris, Inc.), 1993c, *Ports of North Carolina Ports Authority; Wilmington and Morehead City Terminals, Assessment for Receipt of Foreign Research Reactor Spent Nuclear Fuel,* Iselin, NJ, November 23.

Fischer, L. E., C. K. Chou, M. A. Gerhard, C. Y. Kimura, R. W. Martin, R. W. Mensing, M. E. Mount and M. C. Witte, 1987, *Shipping Container Response to Severe Highway and Railway Accident Conditions,* NUREG/CR-4829, UCID-20733, Lawrence Livermore National Laboratory, Livermore, CA.

Flint, R. J., H. Mangues, R. Jenner and M. Garelli, 1993, Director of Operations, Director of Public Safety, Assistant Director of Public Safety, and Chief Hazardous Materials Officer, Port Everglades Authority, Port Everglades, FL, personal communications with C. Miller, Science Applications International Corporation, November 22 and November 23.

FL DEP (Florida Department of Environmental Protection), 1994, Chapter 62-302 Surface Water Quality Standards, Sections 62-302.400, 62-302.600, and 62-302.700, August 8.

Fort Clinch State Park, 1994, Bird List, Amelia Island, FL, January.

Foster, R. J., 1971, *Physical Geology,* Merrill Publishing Company, Columbus, OH.

Frederick, D., 1994, State Supervisor, U.S. Fish and Wildlife Service, Olympia, WA, Letter to D. Scott, Science Applications International Corporation, December 8.

FS (*The Florida Shipper Magazine*), 1992, Special Section, Volume 18, No. 17, December 28.

FWS (U.S. Fish and Wildlife Service), 1992, *Threatened and Endangered Species of Texas,* Texas State Office, Austin, TX, August.

FWS (U.S. Fish and Wildlife Service), 1982a, *Mobile, Gulf Coast Ecological Inventory Map,* 30088-A1-EI-250 (Gulfport, MS).

FWS (U.S. Fish and Wildlife Service), 1982b, *West Palm Beach, Gulf Coast Ecological Inventory Map,* 26080-A1-EI-250 (Port Everglades, FL).

FWS (U.S. Fish and Wildlife Service), 1982c, *Miami, Gulf Coast Ecological Inventory Map,* 25080-A1-EI-250 (Miami, FL).

FWS (U.S. Fish and Wildlife Service), 1982d, *Breton Sound, Gulf Coast Ecological Inventory Map,* 29088-A1-EI-250 (New Orleans, LA).

FWS (U.S. Fish and Wildlife Service), 1981a, *Seattle, Pacific Coast Ecological Inventory Map,* 47122-A1-EI-250 (Seattle, WA and Tacoma, WA).

FWS (U.S. Fish and Wildlife Service), 1981b, *San Francisco, Pacific Coast Ecological Inventory Map,* 37122-A1-EI-250 (San Francisco, CA).

FWS (U.S. Fish and Wildlife Service), 1981c, *Long Beach, Pacific Coast Ecological Inventory Map* (Long Beach and Los Angeles, CA).

FWS (U.S. Fish and Wildlife Service), 1981d, *Vancouver, Pacific Coast Ecological Inventory,* 45122-A1-EI-250 (Portland, OR).

FWS (U.S. Fish and Wildlife Service), 1981e, *Sacramento, Pacific Coast Ecological Inventory Map,* 38120-A1-EI-250 (NWS Concord, CA).

FWS (U.S. Fish and Wildlife Service), 1981f, *Santa Rosa, Pacific Coast Ecological Inventory Map,* 38122-A1-EI-250 (NWS Concord, CA).

FWS (U.S. Fish and Wildlife Service), 1981g, *Los Angeles, Pacific Coast Ecological Inventory Map,* 34118-A1-EI-250.

FWS (U.S. Fish and Wildlife Service), 1981h, *Victoria, Pacific Coast Ecological Inventory Map,* 48122-A1-EI-250.

FWS (U.S. Fish and Wildlife Service), 1980a, *Beaufort, Atlantic Coast Ecological Inventory Map,* 34076-A1-EI-250 (Wilmington, NC).

FWS (U.S. Fish and Wildlife Service), 1980b, *James Island, Atlantic Coast Ecological Inventory Map,* 32078-A1-EI-250 (Charleston, SC).

FWS (U.S. Fish and Wildlife Service), 1980c, *Savannah, Atlantic Coast Ecological Inventory Map,* 32080-A1-EI-250 (Savannah, GA).

FWS (U.S. Fish and Wildlife Service), 1980d, *Norfolk, Atlantic Coast Ecological Inventory Map,* 36076-A1-EI-250 (Hampton Roads, VA).

FWS (U.S. Fish and Wildlife Service), 1980e, *Jacksonville, Atlantic Coast Ecological Inventory Map,* 30080-A1-EI-250 (Jacksonville, FL and Fernandina Beach, FL).

FWS (U.S. Fish and Wildlife Service), 1980f, *Wilmington, Atlantic Coast Ecological Inventory Map,* 39074-A1-EI-250 (Wilmington, DE and Philadelphia, PA).

FWS (U.S. Fish and Wildlife Service), 1980g, *Baltimore, Atlantic Coast Ecological Inventory Map,* 39076-A1-EI-250 (Baltimore, MD).

FWS (U.S. Fish and Wildlife Service), n.d.a, *National Wetlands Inventory Map,* Galveston, TX.

Gaines, E., 1994, Oregon National Heritage Program, Letter to D. Scott, Science Applications International Corporation, April 12.

GNS (General Nuclear Systems, Inc.) 1993, *Safety Analysis Report, Vitrified High-Level Waste Type B Shipping Cask,* Columbia, SC, August.

Goldman, L., 1994, Field Supervisor, U.S. Fish and Wildlife Service, Daphne, AL, Letter to D. Scott, Science Applications International Corporation, November 23.

Gordon, K. L., 1994, Mississippi Natural Heritage Program, Mississippi Department of Wildlife, Fisheries and Parks, Letter to D. Scott, Science Applications International Corporation, September 23.

GPA (Georgia Ports Authority), 1994, Port of Savannah Brochure, Savannah, GA.

Gregory, J. J., N. R. Keltner and R. Mata, 1989, "Thermal Measurements in Large Pool Fires," *Heat Transfer,* Volume 111, May.

Gregory, J. J., R. Mata, Jr. and N. R. Keltner, 1987, *Thermal Measurement in a Series of Large Pool Fires,* SAND85-0196, Sandia National Laboratories, Albuquerque, NM, August.

Hachey, J. and R. Korvola, 1994, General Manager of Marine Operations and Manager of Environmental Sciences Division (Engineering Services), Port of Portland, OR, personal communication with R. L. Gotchy, Science Applications International Corporation, June 22.

Hamilton, W., 1976, *Plate Tectonics and Man,* U.S. Geological Survey Annual Report, Fiscal Year 1976.

Hennessy, R., 1993, Environmental Affairs, New Jersey Terminals, Port Authority of New York and New Jersey, personal communication with C. Miller, Science Applications International Corporation, October 15.

Hilliard, H., 1993, Marketing Manager, Port of Long Beach, CA, personal communication with C. Miller, Science Applications International Corporation, October 19.

Horan, J. P., 1993, Director of Operations, Port of Houston, TX Authority (HPA), personal communication with C. Miller, Science Applications International Corporation, November 19.

Horning, D., 1994, Endangered Species Coordinator, U.S. Fish and Wildlife Service, Raleigh, NC, Letter to D. Scott, Science Applications International Corporation, November 4.

Humphreys, S. L., et al., 1994, *Sector Population, Land Fraction, and Economic Estimation Program (SECPOP90),* SAND93-4032, Sandia National Laboratories, Albuquerque, NM, July, draft.

IAEA (International Atomic Energy Agency), 1990, *Explanatory Material for the IAEA Regulations for the Safe Transport of Radioactive Materials,* Safety Series No. 7, Vienna, Austria.

IAEA (International Atomic Energy Agency), 1961, *Regulation for the Safe Transport of Radioactive Materials,* Notes on Certain Aspects of the Regulations, Safety Series No. 7, Vienna, Austria.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection, ICRP Publication 60,* Volume 21, Pergamon Press, New York, NY, November.

IMO (International Maritime Organization), 1992, Analysis of Fire Casualty Records, FP 37/5, Sub-Committee of Fire Protection, 37th Session, Agenda Item 5, February 20.

IPA (Information Please Almanac), 1993, The 1993 Information Please Almanac, Atlas, and Yearbook, 46th edition, Houghton Mifflin Co., Boston, MA.

Jane's (Jane's Information Group, Inc.), 1992, *Jane's Containerization Directory, 1992-93*, P. Hicks, Editor, Alexandria, VA.

Johnson, C., 1995, Supervisor - South Florida Ecosystem Office, U.S. Fish and Wildlife Service, Vero Beach, FL, Letter to D. Scott, Science Applications International Corporation, April 12.

Jow, H. N., J. L. Sprung, J. A. Rollstin, L. T. Ritchie and D. I. Chanin, 1990, *MELCOR Accident Consequences Code System (MACCS), Model Description,* NUREG/CR-4691, SAND86-1562, Sandia National Laboratories, Albuquerque, NM, February.

Keltner, N. R., et al., 1994, "Simulating Fuel Spill Fires Under the Wing of an Aircraft", *4th International Symposium on Fire Safety Science,* Ottawa, Canada, June.

Kobetich, G., 1994, Field Supervisor, U.S. Fish and Wildlife Service, Carlsbad, CA, Letter to D. Scott, Science Applications International Corporation, November 16.

Kurkoski, D., 1994, U.S. Army Corps of Engineers, Portland District, OR, personal communication to D. Scott, Science Applications International Corporation, with Excerpts from Draft Environmental Assessment, Oregon Slough, Oregon, Channel Dredging, Portland, OR.

Laumeyer, P., 1994, Field Supervisor, U.S. Fish and Wildlife Service, Brunswick, GA, Letter to D. Scott, Science Applications International Corporation, November 16.

Leong, J., 1993, Port Publications, Port of Los Angeles, CA, personal communications with C. Miller, Science Applications International Corporation, October 18 and 19.

Lewis, B., 1995, Naval Weapons Station Charleston, Charleston, SC, personal communication with G. DeMoss, Science Applications International Corporation, October 26.

Lloyd's (Lloyd's Maritime Information Services), 1991, *Casualty File Manual,* London, England, November 18.

Ludwigsen, J. S. and D. Ammerman, 1995, *Extra-Regulatory Impact Tests and Analyses of the Structural Evaluation Test Unit,* (Draft), Sandia National Laboratories, Albuquerque, NM, March.

Magness, M., 1993, Manager of Marine Market Development, Port of Portland, OR, personal communication with C. Miller, Science Applications International Corporation, October 18.

Mayne, K., 1994, Supervisor - Virginia Field Office, U.S. Fish and Wildlife Service, White Marsh, VA, Letter to D. Scott, Science Applications International Corporation, November 2.

McBee, A., 1994, South Carolina Wildlife and Marine Resources Department, Columbia, SC, Letter to D. Scott, Science Applications International Corporation, April 5.

McLendon, R., 1994, Security Chief, Port of Tacoma, WA, personal communication with R. L. Gotchy, Science Applications International Corporation, June 24.

Medlin, J., 1994, Field Supervisor, U.S. Fish and Wildlife Service, Sacramento, CA, Letter to D. Scott, Science Applications International Corporation, November 21.

Menta, J., 1993, Philadelphia Regional Port Authority, personal communication with C. Miller, Science Applications International Corporation, October.

Mets, M., 1994, Deputy Port Director - Operations and Marine Services, Port of Palm Beach, FL, personal communication with C. Miller, Science Applications International Corporation, August 18.

Millar, M., 1995, LCDR, U.S. Coast Guard, Captain of the Port Office, Charleston, SC, personal communication with G. DeMoss, Science Applications International Corporation, September.

Minorsky, V., 1959, "An Analysis of Ship Collisions with Reference to Protection of Nuclear Power Plants," *Journal of Ship Research,* Volume 3, October.

Mitchell, C., 1994, Chief Wharfinger, Port of San Francisco, CA, personal communication with R. L. Gotchy, Science Applications International Corporation, June 24.

Mitchell, C., 1993, Chief Wharfinger, Port of San Francisco, CA, personal communication with C. Miller, Science Applications International Corporation, October 25.

Moise, A. M., W. D. Welch and J. Bryant, 1993, Director of Public Relations, Executive Director, and Senior Director of Operations and Engineering, South Carolina State Ports Authority, Port of Charleston, SC, personal communication with C. Miller, Science Applications International Corporation, October 14.

Moore, T. L., 1993, Director of Public Relations, Georgia Ports Authority, Port of Savannah, GA, personal communication with C. Miller, Science Applications International Corporation, October 14.

Moriconi, R., 1993, Deputy Port Director for Operations, Massachusetts Port Authority, Maritime Department, personal communication with C. Miller, Science Applications International Corporation, October 18.

Motora, S., 1971, *J. of Japanese Soc. Naval Architec.*, Volume 7.

MTMC (Military Traffic Management Command), 1994a, personal communication to R. L. Gotchy, Science Applications International Corporation, August 9.

MTMC (Military Traffic Management Command), 1994b, *Military Traffic Management Command Western Area Commander's Pocket Digest,* FY 1994, 2nd Quarter.

MTMCTEA (Military Traffic Management Command, Transportation Engineering Agency), 1992, *Ports for National Defense; An Analysis of Unit Deployments through U.S. East Coast Ports,* SE 90-3d-21, Newport News, VA, June.

MTMCTEA (Military Traffic Management Command, Transportation Engineering Agency), 1990, *Ports for National Defense; An Analysis of Unit Deployments through U.S. Ports,* SE 89-3d-31, Newport News, VA, October.

Mudan, K. S. and P. A. Croce, 1988, "Fire Hazard Calculations for Large Open Hydrocarbon Fires", *SPFE Handbook of Fire Protection Engineering,* P. J. DiNenno, Editor, National Fire Protection Association, Quincy, MA.

Murray, S., 1994, Florida Natural Areas Inventory, Letter to D. Scott, Science Applications International Corporation, April 27.

NCDEHNR (North Carolina Department of Environment, Health, and Natural Resources), 1992, *Water Quality Progress in North Carolina 1990-1991 305(b) Report,* Report No. 92-06, Division of Environmental Management-Water Quality Section, September.

NCSPA (North Carolina State Ports Authority), 1994, Various North Carolina Port Brochures including: *1994 Services Directory*; *Wilmington; Morehead City*; *1992 Economic Impact Study Summary*; and other related information; Wilmington, NC.

NDAA (National Defense Authorization Act for Fiscal Year 1994), 1993, Public Law 103-160, Section 3151, *Limitations on Receipt and Storage for Spent Nuclear Fuel from Foreign Research Reactors,* Washington, DC, November 30.

Nelsen, J. M., 1986, *Determination of Net Heat Fluxes and Assessment of Test-to-Test Thermal Input Variation for Three Large Engulfing Open Pool Fire/Calorimeter Tests,* SAND85-1821, Sandia National Laboratories, Albuquerque, NM, November.

NOAA (National Oceanic and Atmospheric Administration), 1993a, *Local Climatological Data: Annual Summary with Comparative Data - Baltimore, MD,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993b, *Local Climatological Data: Annual Summary with Comparative Data - Miami, FL,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993c, *Local Climatological Data: Annual Summary with Comparative Data - Galveston, TX,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993d, *Local Climatological Data: Annual Summary with Comparative Data - San Francisco, CA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993e, *Local Climatological Data: Annual Summary with Comparative Data - Los Angeles, CA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1993f, *Local Climatological Data: Annual Summary with Comparative Data - Long Beach, CA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992a, *Local Climatological Data: Annual Summary with Comparative Data - Norfolk, VA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992b, *Local Climatological Data: Annual Summary with Comparative Data - Wilmington, NC,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992c, *Local Climatological Data: Annual Summary with Comparative Data - Charleston, SC,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992d, *Local Climatological Data: Annual Summary with Comparative Data - Savannah, GA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992e, *Local Climatological Data: Annual Summary with Comparative Data - Jacksonville, FL,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992f, *Local Climatological Data: Annual Summary with Comparative Data - Portland, OR,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992g, *Local Climatological Data: Annual Summary with Comparative Data - Seattle, WA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992h, *Local Climatological Data: Annual Summary with Comparative Data - Philadelphia, PA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992i, *Local Climatological Data: Annual Summary with Comparative Data - Mobile, AL,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992j, *Local Climatological Data: Annual Summary with Comparative Data - Stockton, CA,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992k, *Local Climatological Data: Annual Summary with Comparative Data - Wilmington, DE,* National Climatic Data Center, Asheville, NC.

NOAA (National Oceanic and Atmospheric Administration), 1992l, *Local Climatological Data: Annual Summary with Comparative Data - New Orleans, LA,* National Climatic Data Center, Asheville, NC.

Noland, J., 1994, Seattle City Council, Letter (with attachments) to D. Lillian, Department of Energy, Office of Technology Development, January 20.

NRC (U.S. Nuclear Regulatory Commission), 1993, *Public Information Circular for Shipments of Irradiated Reactor Fuel,* NUREG-0725, Rev. 9, Washington, DC.

NRC (U.S. Nuclear Regulatory Commission), 1990, *Packaging and Transportation of Radioactive Material,* 10 CFR 71.

NRC (U.S. Nuclear Regulatory Commission), 1988, *Packaging and Transportation of Radioactive Materials,* Title 10 Part 71, Subpart F, Packaging and Special Form Tests, Section 71.73, Hypothetical Accident Conditions.

NRC (U.S. Nuclear Regulatory Commission), 1977, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes,* NUREG-0170, Volumes I and II, Washington, DC.

O'Connell, T., 1994, Virginia Department of Conservation and Recreation, Division of Natural Heritage, Richmond, VA, Letter to D. Scott, Science Applications International Corporation, March 25.

ORI, 1981a, King, P., J. Kirkland and B. Paramore, *Hazardous Environments Experienced by Radioactive Material Packages Transported by Water,* Silver Spring, MD, draft, November.

ORI, 1981b, Kirkland, J. and T. Baranano, *Accident Severities Experienced by Radioactive Material Packages Transported by Water,* Silver Spring, MD, draft, November.

Parker, K., L. Spalluto and J. Cefalu, 1993, Manager for Port Safety and Security, Port-Wide Maintenance Manager, and Superintendent of Marine Terminals, Port of New Orleans, LA, personal communication with C. Miller, Science Applications International Corporation, November 18.

Paulsen, L., 1994, Director of Risk Management, Port of Tacoma, WA, personal communication with R. L. Gotchy, Science Applications International Corporation, June 21.

Paulsen, L., 1993, Director of Risk Management, Port of Tacoma, WA, personal communication with C. Miller, Science Applications International Corporation, October 19.

PEA (Port Everglades Authority), 1993, Guide to Port Everglades, Hollywood/Ft. Lauderdale/Dania, 1992-1993, Alson Marketing, Inc., Fort Lauderdale, FL; and Port Report, Spring/Summer 1993, PEA, Fort Lauderdale, FL.

Perry, E. W., 1994, U.S. Fish and Wildlife Service, State College, PA, Letter to D. Scott, Science Applications International Corporation, November 2.

Petersen, M., 1982, "Dynamics of Ship Collisions," *Ocean Engineering,* Volume 9, No. 4.

PNDI (Pennsylvania Natural Diversity Inventory Information System), 1994, Pennsylvania Department of Environmental Resources, Bureau of Forestry, Harrisburg, PA, personal communication to D. Scott, Science Applications International Corporation, June 14.

POB (Port of Boston), 1993, *The Port of Boston: New England's International Gateway,* Port Brochure and Marketing Materials, Massachusetts Port Authority, Boston, MA.

POHA (Port of Houston Authority), 1993, *The Houston Ship Channel, Economic Lifeline to the World,* Houston, TX.

POLA (Port of Los Angeles), 1994, Various Port Brochures including: *1993 Los Angeles Annual Report*; *1994 WORLDPORT LA Shipping Handbook*; *Port of Los Angeles Tariff No. 4, Section Seventeen (Dangerous Cargo and Explosives on Vessels)*; and related information; Los Angeles, CA; personal communication to R. L. Gotchy, Science Applications International Corporation, June 23.

POLB (Port of Long Beach), 1993a, *A Quick Guide to the Port of Long Beach,* Long Beach, CA.

POLB (Port of Long Beach), 1993b, *Harbor at a Glance,* Long Beach, CA.

POLB (Port of Long Beach),1993c, *Harbor Handbook,* a Digest of Port Facilities and Services, Long Beach, CA.

POLB (Port of Long Beach), 1993d, *Interport,* Long Beach, CA, summer.

PON (Port of New Orleans), 1994, Various Port Brochures including: *Port of New Orleans 1993-1994 Annual Directory*; *Highway & Harbor Map*; *Port of New Orleans Record* (various issues); *Mississippi River Terminal Complex*; and related information; Port Authority of Greater New Orleans, LA.

POP (Port of Portland), 1994, Various Port Brochures including: *Marine Facilities Guidelines*; *Your World Port - Oregon's Gateway for Business*; *Portside* (Summer 1994); Port of Portland, OR.

POPB (Port of Palm Beach), 1994, Various Port Brochures including: *1993 Annual Report*; *Port of Palm Beach District General Information - 1994*; *Gateway* (various issues); Port of Palm Beach, Riviera Beach, FL; personal communication to C. Miller, Science Applications International Corporation, August 18.

PORT (Port of Richmond Terminal), 1994, Various Port Brochures including: 1993 Annual Report; The Port of Opportunity; Port of Richmond Fact Sheet; Shipping News; Port News; 1993 Tonnage Summary; Marine Terminal Facilities; Port of Richmond, VA.

POS (Port of Seattle), 1994, Various Port Brochures including: *1993 Annual Report*; *1992 Annual Report*; *1991 Facilities Handbook*; *Tradelines* (various issues); and related information; Port of Seattle, WA.

POSF (Port of San Francisco), 1993, Welcome to the Port of San Francisco, San Francisco, CA.

POT (Port of Tacoma), 1994, Various Port Brochures including: *1993 Annual Report*; *Facilities and Services (15M/11-92)*; *Facilities & Services Summary (15M/9-91)*; *Truck on In*; *Pacific Gateway* (winter 1994); Port of Tacoma, WA.

POW (Port of Wilmington), 1994, The Port of Personal Services brochure, Port of Wilmington Profile, and *Port Illustrated* (various issues), Port of Wilmington, DE.

Powell, W. D., G. Knatz, W. Wilson and T. Johnson, 1994, Chief Wharfinger, Director of Planning, Director of Security, and Environmental Specialist, Port of Long Beach, CA, personal communication with R. L. Gotchy, Science Applications International Corporation, June 23.

Powers, D. A., L. N. Kmetyk and R. C. Schmidt, 1994, *A Review of the Technical Issues of Air Ingression During Severe Reactor Accidents,* NUREG/CR-6218, SAND94-0731, Sandia National Laboratories, September.

PT (Penn Terminals), 1994, Penn Terminals Inc. Brochure, Eddystone, PA, undated.

Reaves, R., 1994, Mississippi Department of Environmental Protection, personal communication with D. Scott, Science Applications International Corporation, August 29.

Richards, T., 1994, U.S. Fish and Wildlife Service, personal communication with D. Scott, Science Applications International Corporation, November 17.

Robas, V., 1994, Director, Port Administration, Port of Fernandina, FL, Letter to D. Amick, Science Applications International Corporation, November 29.

Sanders, T. L., K. D. Seager, Y. R. Rashid, P. R. Barnett, A. P. Malinauskas, R. E. Einziger, H. Jordan, T. A. Duffey, S. H. Sutherland and P. C. Reardon, 1992, *A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements,* SAND90-2406, Sandia National Laboratories, Albuquerque, NM, November.

Schmitt, D. N., 1993, *Savannah and Ogeechee River Creek Surveys,* Georgia Department of Natural Resources, Game and Fish Division, Richmond Hill, GA.

Schneider, M. E., N. R. Keltner and L. A. Kent, 1989, *Thermal Measurements in the Nuclear Winter Fire Test,* SAND88-2839, UC-722, Sandia National Laboratories, Albuquerque, NM, November.

Schultz, R. C., 1993, Director of Operations, Port of Galveston, TX, personal communication with C. Miller, Science Applications International Corporation, November 19.

Scott, J., 1994, Executive Director, Port of Wilmington, NC, personal communication with D. Amick, Science Applications International Corporation, August 9.

Shaffer, C. S., 1994, Shipping Cask Transportation Accident MELCOR Calculations, Seminar Presentation, Science and Engineering Associates, Inc., prepared for Sandia National Laboratories, Albuquerque, NM, November 18.

Silver, R. M., 1995, Director of Operations, Military Traffic Management Command at North Charleston, SC, personal communication to Pat Wells, U.S. Department of Energy, September 27.

Smith, I., 1994, North Carolina Department of Environment, Health, and Natural Resources, Raleigh, NC, Letter to C. Cross, Science Applications International Corporation, April 7.

SNL (Sandia National Laboratories), 1994, Science Applications International Corporation Query of Radioactive Materials Postnotification (RAMPOST) Database, April 6.

SNL (Sandia National Laboratories), 1989, *Beneficial Uses Shipping System Cask (BUSS) Safety Analysis Report for Packaging (SARP),* SAND83-0698, Albuquerque, NM, March.

Southern Shipper, 1993, *Southern Shipper, Southern Ports 1993*, An American Shipper Publication, Jacksonville, FL, January.

Sprung, J. L., J. A. Rollstin, J. C. Helton and H. N. Jow, 1990, *Evaluation of Severe Accident Risks: Quantification of Major Input Parameters, MACCS Input,* Volume 2, Rev. 1, Part 7, NUREG/CR-4551, SAND86-1309, Sandia National Laboratories, Albuquerque, NM, December.

SRI (SRI International), 1978, *Cost-Effectiveness of Marine Fire Protection Programs,* Prepared for the U.S. Department of Commerce, Maritime Administration, and the National Bureau of Standards, November.

Stark, T. B., 1995, Captain U.S.N., Commander of Naval Weapons Station at Charleston, SC, personal communication with K. Chacey, Department of Energy, Office of Spent Nuclear Fuel Management, Washington, DC, September 18.

State of California, 1986, *Basin Plan,* California Regional Water Quality Control Board, Oakland, CA.

Stubbs, W., 1994, President, Nassau County Ocean Highway and Port Authority, Port of Fernandina, FL, personal communication with C. Miller, Science Applications International Corporation, August 30.

TPWD (Texas Parks and Wildlife Department), 1989a, *Trends in Texas Commercial Fishery Landings, 1977-1988,* Fisheries Division, Management Data Series No. 7, Austin, TX.

TPWD (Texas Parks and Wildlife Department), 1989b, *Trends in Finfish Landings by Sport-boat Fishermen in Texas Marine Waters, May 1974-1988,* Fisheries Division, Management Data Series No. 8, Austin, TX.

UBC (Uniform Building Code), 1991, *International Conference of Building Officials,* Whittier, CA.

U.S. Army (U.S. Army Corps of Engineers), 1994, San Francisco District, *Final Supplemental Environmental Impact Report/Environmental Impact Statement Oakland Harbor Deep-Draft Navigation Improvements,* June.

U.S. Army (U.S. Army Corps of Engineers), 1993, *Environmental Assessment and Finding of No Significant Impact, Wilmington Harbor Ocean Bar Channel Deepening, Wilmington, North Carolina,* Wilmington District, June.

U.S. Army (U.S. Army Corps of Engineers), 1991, *Environmental Impact Study, Savannah Harbor, Georgia Comprehensive Study,* Savannah District, July.

U.S. Army (U.S. Army Corps of Engineers), 1990, *Draft Environmental Impact Statement/Environmental Impact Report Phase I 2020 Plan and Feasibility Study (Channel Improvements and Landfill Development),* Los Angeles District, and Long Beach and Los Angeles Harbor Departments, September.

USCG (U.S. Coast Guard), 1994a, Office of Marine Safety, Security and Environmental Protection, Washington, DC, personal communication to D. Amick, Science Applications International Corporation, November 3.

USCG (U.S. Coast Guard), 1994b, Shipping Accident Survey Data, U.S. Coast Guard Offices (Marine Safety Division, Washington, DC; Marine Safety Office, Portland, OR; Marine Safety Office, San Francisco, CA; Marine Safety Office, Long Beach/Los Angeles, CA), personal communication to D. Amick, Science Applications International Corporation, August.

U.S. District Court, 1991, Sierra Club v. Energy Department, 808 F.Supp. 852, 34 ERC 2057, District of Columbia, December 9.

USMMA (U.S. Merchant Marine Academy), 1994, *Report on the Workshop on Port Selection Criteria for Shipments of Foreign Research Reactor Spent Nuclear Fuel, Kings Point, NY* (November 15-16, 1993), Sandia National Laboratories, Albuquerque, NM.

Verhoef, R. W., E. Paz, T. L. Garrett and J. Leong, 1994, Chief Wharfinger, Hazardous Materials Investigator, Environmental Scientist, and Public Relations Specialist, Interviews with R. L. Gotchy, and D. Amick, Science Applications International Corporation, June 23.

VPA (Virginia Port Authority), 1994, Various Port Brochures including: *Promises, Results, The Ports of Virginia; The Ports of Virginia - One Stop, America and the World; Virginia Maritimer* (July/August 1994); and other related information; VPA, Norfolk, VA.

Warwick, J. E. and A. L. Anderson, 1976, *The Nature of Ship Collisions Within Ports,* Todd Shipyard Corp., Galveston, TX (prepared for the U.S. Maritime Administration), April.

Wayland, R. J. and S. Raman, 1989, "Mean and Turbulent Structure of a Baroclinic Marine Boundary Layer During the 28 January 1986 Cold Air Outbreak (GALE 86)", *Boundary-Layer Meteorology,* Volume 48, No. 3, August.

WDW [Washington (State) Department of Wildlife], 1994a, *Priority Habitats and Species Map, Seattle South,* No. 4712253, April 13.

WDW [Washington (State) Department of Wildlife], 1994b, *Priority Habitats and Species Map, Tacoma North,* No. 4712234, April 13.

Werner, F. T., 1994, U.S. Fish and Wildlife Service, Division of Ecological Services, Houston, TX, Letter to D. Scott, Science Applications International Corporation, November 2.

West, T., 1994, Virginia Department of Environmental Quality, Virginia Beach, VA, Letter (with water quality data) to D. Scott, Science Applications International Corporation, April 15.

Wilmot, E. L., 1981, *Transportation Accident Scenarios for Commercial Spent Fuel,* SAND80-2124, Sandia National Laboratories, Albuquerque, NM, February.

Wilmot, E. L., J. D. McClure and R. E. Luna, 1981, *Report on a Workshop on Transportation Accident Scenarios involving Spent Fuel,* SAND80-2012, TTC-0151, Transportation Analysis and Information Division, Sandia National Laboratories, Albuquerque, NM, February.

Wilson, J. L., 1995, Assistant to the Executive Director, North Carolina State Ports Authority, Port of Wilmington, NC, personal communications with R. L. Gotchy, Science Applications International Corporation, September 6 and 7.

Wilson, J. L., 1993, Assistant to the Executive Director, North Carolina State Ports Authority, Port of Wilmington, NC, personal communication with C. Miller, Science Applications International Corporation, October 14.

Wolflin, J., 1994, Supervisor, Chesapeake Bay Field Office, U.S. Fish and Wildlife Service, Annapolis, MD, Letter to D. Scott, Science Applications International Corporation, November 7.

World Almanac, 1992, *World Almanac and Book of Facts,* Pharos Books, New York, NY, p. 940.

Yocum, K., 1994a, Development Manager, Concord NWS, CA, personal communication with R. L. Gotchy, Science Applications International Corporation, August 15 and 16.

Yocum, K., 1994b, Strategic Planning Officer, Concord NWS, CA, personal communication to R. L. Gotchy, Science Applications International Corporation, September 1.

Attachment D1
Capital Improvement Plans and Other Significant Port Developments Anticipated During the Period of the Proposed Action

Port	*Improvements and Other Significant Developments*
Baltimore, MD	The Board of Public Works approved the purchase of a new $7.4 million container crane to be installed at the Dundalk Marine Terminal, a 231 ha (570 acre) terminal complex with 13 deepwater berths and 9 container cranes. The new crane is expected to be operational by early 1995 (Governors Press Office, State of Maryland, May 18, 1994). Governor William Schaefer announced Board of Public Works approval of a contract to modernize and improve (up-grade to post-Panamax capacity) three container cranes located at the Dundalk Marine Terminal (Ibid, June 22, 1994).
Boston, MA	Officials of the Massachusetts Port Authority (Massport) have submitted a draft environmental impact report to Federal and State officials that calls for dredging the harbor and access channels to 12.2 m (40 ft) from 9.75-10.97 m (32-36 ft) (*American Shipper*, "Boston Seeks Direct Calls From Asia," October 1994, Pg. 94). Ralph Cox, Marine Director, and other port officials claim that the deeper water is critical to the Port's viability. Massport is also seeking support of the State Legislature for road and rail clearances to permit double-stack train service to the City of Boston and its marine terminals. A $50 million expansion and modernization of Boston's Conley Terminal is approximately 80 percent complete. When completed, Conley Terminal will have 40.5 ha (100 acres) of container storage and handling area, 4 post-Panamax container cranes, 304.8 m (1,000 ft) of berth, and a new gate complex. Reportedly, container tonnage is up for 1994 over 1993 tonnage when Boston handled 152,240 twenty-foot equivalent units for the year.
Charleston, SC	As of late 1993, the $90 million Wando terminal expansion project was nearing completion. When completed, the project will add an additional 418.5 m (1,373 ft) of berthing space, 26.3 ha (65 acres) of container storage area, and two $5.4 million post-Panamax container cranes. The entire project is scheduled for completion by Fall 1994. Planning is progressing for development of the approximately 323.8 ha (800-plus acre) Daniel Island terminal site in Charleston Harbor. The container terminal is being designed to meet demand at the port well into the 21st century. The massive project is expected to take 15 to 20 years to complete and will ultimately consist of 323.8 ha (800 acres) of paved container storage and 2,438 m (8,000 lineal ft) of berthing space (*American Shipper* - Southern Ports, January 1994). At NWS Charleston, the U.S. Army is planning to expand Wharf Alpha and upgrade the railroad in support of the Army Strategic Mobility Logistics Base. This upgrade is scheduled for completion in 1998.

Concord NWS, CA	Currently authorized improvements expected to be completed by 1997 include an upgrade of Pier 3 to withstand greater loadings, and will also include two new 36 metric ton (40 ton) container cranes, container storage pads, and support facilities and equipment. These improvements are projected to permit average load rates of about 20 containers per hour. The improvements will also permit increasing the channel depth and depth alongside to 12.7 m (42 ft) in the future if necessary. The facility will be the designated West Coast container facility for military shipments (personal communications from Karl Yocum, Concord NWS Office of Business Development, September 1, 1994, and 1994 Fact Sheets received during port visit).
Eddystone, PA	No immediate improvements identified.
Fernandina Beach, FL	U.S. Army Corps of Engineers is expected to award a contract in October 1994 for deepening of the harbor channel to 11 m (36 ft), and to construct a 366 m (1,200 ft) turning basin (Personal communication from Mr. Stubbs, Port of Fernandina, September, 1994).
Gulfport, MS	A consulting firm has recommended a 15-year, $160 million terminal expansion for the Port of Gulfport to a projected tripling of port business by 2010. The proposed expansion would add 34 ha (84 acres) of land to West Pier at a cost of $81 million, replace or reconfigure existing warehouses ($13 million), and include the purchase of $11 million in additional container handling equipment (but not necessarily a container gantry crane) (*American Shipper*, September 1994, p. 102; *Containerization International*, September 1994, p. 11). A $41 million dredging project that deepened the harbor from 9.14 m to 10.97 m (30-36 ft) was completed in April 1994. In August 1993, the Port Authority issued $15 million in bonds to pay for three development projects that included expansion of East Pier warehouse facilities and the addition of 11.7 ha (29 acres) of land—through diking and pumping sand, at West Pier. The latter to be used for a new container terminal (*American Shipper* - Southern Ports, January 1994).
Jacksonville, FL	In anticipation of continued strong growth in cargo demand over the next 20 years, JAXPORT adopted a 20-year, $934 million development plan designed to prepare its facilities for 2010. In addition to recommendations for immediate construction of a third container terminal at Dames Point, consultants recommended expansion and reconfiguration of the Authority's Blount Island and Talleyrand Terminals, projected to cost about $274 million over 7 years. A $1 million feasibility cost sharing agreement was signed this year with the Corps of Engineers to develop a dredging study to deepen the harbor from 11.6 m (38 ft) to 12.8 m (42 ft). Design of a second roll-on/roll-off dock plus 2 ha (5 acres) more container storage and a 450 m (1,500 ft) extension of marginal wharf is scheduled for Blount Island in fiscal 1995 [The Jacksonville Port Authority (JAXPORT), Marketing Department, October 3, 1994].

The Port Authority anticipated breaking ground in 1994 for a new 203 ha (500 acre) container and general cargo terminal complex at Dames Point, immediately adjacent and upstream of its existing Blount Island terminal. The new facility is expected to cost $160 million when completed in the future. Other improvements scheduled include a $2.5 million investment in increased intermodal rail capacity at Blount Island and Talleyrand Terminals, and widening of Hecksher Drive to four lanes from the entrance to Blount Island to State Road 9A, which connects with I-95. I-295 is also being widened to four lanes (American Shipper - Southern Ports, January 1994).

Long Beach, CA

The Long Beach Port Commission and City Council have approved a 1994-95 budget of $417 million, which includes $236.5 million for port construction, land acquisition, and environmental mitigation. Last year's budget included $405 million to purchase land owned by Union Pacific Resources Company in the north harbor, which the Port plans to convert to a marine cargo terminal. The property is comprised of 117 ha (289 acres) north of the Cerritos Channel, 143 ha (354 acres) south of the Channel, and 33 ha (82 acres) within the Channel. The new budget provides allocations of $60 million for street overpasses to cross rail lines in the port area, $22 million for other street and road improvements, $78 million for continuing container terminal improvements at Pier J, $25 million for other construction projects, and $40 million for land acquisitions and environmental mitigation. These land acquisitions will increase the Port's operating area by 35 percent (American Shipper, September 1994, p. 94; "Long Beach to Spend $417 million").

Los Angeles, CA

Los Angeles' "2020 Program" represents the Port's comprehensive long-term development plan, which is designed to accommodate a doubling of cargo throughput through the next decade and a forecast California population of 20 million people. The major components of the 2020 Program include:

a. Construction of Pier 300 on landfill completed in 1983. When completed, Pier 300 will include the American President Lines container terminal, an intermodal container/rail/truck transfer facility and a coal export terminal;

b. Landfill and construction of Pier 400, with three container terminals, an intermodal container transfer facility, and liquid bulk terminals;

c. The Alameda Corridor, a road and rail improvement program linking the Port to rail facilities in downtown Los Angeles with a fully grade-separated trackage (Port of Los Angeles, Property Management Division, October 3, 1994).

Implementation of the 2020 Program is well underway and will involve expenditures of approximately $600 million over the next three years. Work has begun on the new 91.5 ha (226 acres) American President Lines Container Terminal on Pier 300 which, when completed in 1997, will be the largest container terminal in the United States. Costing about $270 million, the terminal will have 1,219 m (4,000 ft) of wharf capable of handling four of American President Lines' largest ships at one time, and an adjacent 19 ha

(47 acre) intermodal rail yard that will also serve the coal export yard being constructed next to the facility. The American President Lines complex will be equipped with six to eight new-generation container cranes. The Port has also embarked on a mammoth $148.6 million dredging project that will create 4.8 km (3 mi) of new channels 13.6 m to 19.1 m (45-63 ft) deep, providing access to Pier 300, a turning basin, and 1,520 m (5,000 ft) of berthing space south of Pier 300. Dredge spoil will be used to create about 91 ha (225 acres) of new land to be called Pier 400, which will be located south (seaward) of the new American President Lines Terminal. Plans for Pier 400, call for the construction of three container terminals on the north side of the terminal, each with two berths and five container gantry cranes, and a large bulk liquid/petroleum terminal complex on the south (ocean) side. Other on-going improvement projects include replacement of the Badger Avenue Bridge providing rail and road access to Pier 300 and Terminal Island, construction of a $200 million coal export terminal on Pier 300 and a $20 million rail yard to serve Terminal Island container terminals [*Long Beach Press Telegram* (Business), "Port Builds for Future," September 26, 1994].

Miami, FL

Phase I of Miami's $100 million port deepening project (begun in April 1991), was completed July 1993 and included deepening of the harbor channel to 12.8 m (42 ft) from the sea buoy to the Lummus Island Container Terminal. Phase II (now underway) extends the 12.8 m (42 ft) channel from container berths on Lummus Island to a new south channel turning basin between Dodge and Lummus Islands. Completion of dredging is expected by mid-1995. The dredging project has already added 24.3 ha (60 acres) of land to Lummus Island and current dredging is expected to add another 16.2 ha (40 acres) for additional container and roll-on/roll-off ship berths. The Port also plans to add two 49 metric ton (54 ton) post-Panamax size container cranes to the existing three 49 mt and three 39.2 metric ton (43 ton) gantry cranes already installed (American Shipper-Southern Ports, January 1994).

Mobile, AL

The Port has just completed about $80 million in improvements through 1993. No new immediate improvements have been identified (Alabama State Docks System, "Port of Mobile Handbook," 1993).

New Orleans, LA

The newest terminal to be added to the Port of New Orleans is the Nashville "B" multi-purpose facility, which marked the completion of the first phase in the ongoing $200 million capital improvements program which, as part of the Mississippi River Terminal Complex, will take New Orleans into the 21st century. When complete, it will feature two miles of continuous modern wharves and state-of-the-art facilities. A full array of multipurpose and ocean-going container ships will be able to discharge cargo quickly, take on new cargo and sail for the next port without delay. A newly paved marshalling yard will eliminate trucking congestion and tie-ups, and an increased shedded area will allow stevedores to operate more efficiently. Flood protection barriers are being raised to eliminate the possibility of flooding. Two ship berths have been added and three more are scheduled to open by the end of 1995. The Napoleon Avenue Wharf C apron width will be replaced to increase the load capacity to 36 kg sm (850 psf), along with other improvements. The Tchoupitoulas Corridor Project will provide a new, high-speed dedicated roadway from the port through the city (Annual Directory, Port of New Orleans, 1993-1994; Board of Commissioners of the Port of New Orleans, "Mississippi River Terminal Complex," 1993).

Newport News, VA — No immediate improvements identified.

Norfolk, VA — In 1991, the Virginia Ports Authority began a $40 million expansion of the Norfolk International Terminal that will double the size and cargo handling capacity of the terminal. When completed in 2004, improvements include adding 1,300 m (4,300 ft) of new berthing space and 120 ha (300 acres) of backup cargo handling area, creating a massive (819 acre) intermodal terminal with 27,000 m (89,000 ft) of onsite rail, connecting the terminal with Norfolk Southern's bullet train and providing double stack service to major U.S. markets (Virginia Port Authority, "Promises, Results," 1993; Financial World, "The Ports of Virginia: Destiny Controlled," p. 63, New York, NY, July 20, 1993).

Oakland, CA — The $50 million reconstruction of Oakland's 22.7 ha (56 acre) Seventh Street Terminal is nearing completion. Severely damaged in the 1989 Loma Prieta earthquake, three new post-Panamax cranes have been added and the entire wharf structure and upland areas have been rebuilt. The final phase of the redevelopment program is a $5 million gate relocation and construction project providing six entry and four exit lanes. Truck queues outside the terminal will be avoided by the addition of 46 inbound and 44 outbound queue spaces plus six "trouble" lanes for trucker paperwork problems within the gate area. The gate complex will use computer and video technology to speed container movements through the Port (American Shipper, August 1994, "Rebirth for Oakland Terminal," p. 77).

Philadelphia, PA — A new bi-state agency, *The Port of Philadelphia and Camden, Inc.*, has been created to assume responsibility for regional port operations previously directed by the Philadelphia Regional Port Authority (ports of Philadelphia), the South Jersey Port Corporation (terminals in Camden), and the World Trade Division of the Delaware River Port Authority—a regional economic development agency. The new agency will begin operation in 1995 (WWS/World Wide Shipping, June 1994, p. 35).

Port Everglades, FL — Completion of the Port Everglades Authority's new $100 million, 62.7 ha (155-acre) container complex at Southport, and the development of 6.7 ha (15 acres) of expanded container storage area at Midport, both scheduled for 1994, culminates years of planning and construction by Port Everglades. Southport is equipped with three 39.2 metric ton (43 ton) low-profile, post-Panamax container cranes designed to avoid interference with nearby airport operations. Design planning studies are underway for lift-on/lift-off support facilities at the new 26 ha (63 acre) lift-on/lift-off container yard located immediately adjacent to Southport's cranes. These include a container freight station, electrical outlets for reefer containers, gatehouse with scales, inspection shed, automated facilities, and a feasibility study for developing an intermodal container transfer facility nearer to the Southport complex. The

Fiscal 1993-94 budget provides \$9.6 million for a tenth cruise line terminal and enhancements to the two facilities described above (FS, 1992; Southern Ports, January 1994, Pg. 33).

Port of New York, NY

The Port Authority of New York/New Jersey's 1993 capital spending budget totaled \$57 million, largely for terminal improvements such as wharf rehabilitation, berth deepening, paving, etc.

Port of Elizabeth, NJ

Funds were also included for deepening Federal channels in the Kill Van Kull and into Newark Bay to the Elizabeth Marine Terminal. The total project, scheduled for completion in 1995, will provide a 12.2 m (40 ft) channel from Upper New York Bay through the Kill Van Kull into Newark Bay. The lack of adequate channel depths has resulted in the diversion of ships to other ports. Three and a half years of wrangling over permits for maintenance dredging and ocean spoil disposal have reportedly increased the cost of dredging from \$1 million to \$15 million, in part due to court-ordered dredging requirements. The Port Authority has previously announced that it will construct a new \$8.5 million on-dock rail terminal at its Port Elizabeth container facilities, which is scheduled for completion in the first quarter of 1995. Initial capacity of the facility will be 100,000 containers annually (WWS/World Wide Shipping, June 1994, p. 33).

Red Hook Container Terminal - Brooklyn, Howland Hook Container Terminal - Staten Island, NY. Red Hook terminal is the only marine cargo terminal still operating on the East side of the Harbor. It was reactivated in January 1994 under the management of American Stevedoring Ltd. The NY/NJ Port Authority is in the process of dredging the approach channel to its project depth of 11.6 m (38 ft). American Stevedoring anticipates handling 20,000 twenty-foot equivalent units in 1994 and as many as 70,000 twenty-foot equivalent units by 1995. The terminal also benefits from a Port Authority subsidized container-on-barge service connecting Red Hook with New Jersey railheads. Terminal facilities include 920 m (3,030 ft) of berthing, containers, roll-on/roll-off and breakbulk cargoes, rail service, four 36.3 metric ton (40 tons) container cranes, and one 63.7 metric ton (70 tons) container crane. While seeking an operator to revitalize the 58.7 ha (145 acre) Howland Hook container terminal—the former base of U.S. Lines idled since 1991, the Port Authority is completing a \$25 million renovation of the terminal. Work includes replacement of electrical and distribution systems and resurfacing of a 762 m (2,500 ft) wharf. The Port Authority is also seeking a dredging permit to increase the depth of the berths from the original 10.1 m (33 ft) to the authorized depth of 12.2 m (40 ft). The terminal has a capacity of more than 300,000 containers a year. Its facilities include 762 m (2,500 ft) of lineal berthing space, four 36.3 metric ton (40 ton) and two 45.5 metric ton (50 ton) container cranes, and rail service (American Shipper, August 1994, Pages 73-74).

The City of New York – owner of the Terminal (Howland Hook), and the State of New Jersey are negotiating for the purchase of the Staten Island Railroad tracks between the Terminal and Cranford, NJ, where the short line connects with Conrail. CSX owns the Staten Island line, but was granted approval in 1991 to abandon the route, so a new owner is needed to reactivate the rail line. City officials and the prospective operator of the Howland Hook facility predicted that the future of this terminal as a viable facility may hinge on the acquisition of the trackage and the installation of on-dock rail service (American Shipper, August 1994, p. 84).

Portland, OR

The Port Commission has approved a $60 million container terminal upgrade program for Terminal 6 to increase throughput capacity to 510,000 twenty-foot equivalent units over the next 10 years, nearly double its present capacity. The Terminal currently handles 314,500 twenty-foot equivalent units a year. Improvement plans include a new $16 million post-Panamax size container crane scheduled to come on stream by late 1995. The Port Commission has also hired an engineering consulting firm to recommend a development strategy and 20-year development program for a new marine terminal complex on West Hayden Island (*American Shipper*, October 1994, "Port of Portland Builds for the Future")

In July, the Port Of Portland Commission contracted with IBM and Stevedoring Services of America to provide the hardware and software for a new $1.0 million computerized terminal management system for its Terminal 6 container facility. The Port presently handles 600 trucks a day with a cargo inventory system developed in 1980. Portland is the fastest growing port on the West Coast (Containerization International, September 1994, "Portland Buys SSA System").

Portsmouth, VA

No immediate improvements identified.

San Francisco, CA

San Francisco's future as a leading West Coast container port is in jeopardy following the decision of Evergreen line to leave the port when its lease expires in June 1995. Evergreen's move follows the departure of Cosco, National Shipping Co. of the Philippines, Nedlloyd Line, Blue Star Lines, and South Seas Steamship. The anticipated reduction in revenues caused by these defections to the Port of Oakland may effect San Francisco's Port capital expenditure programs, including the $10 million rail tunnel improvement project designed to accommodate double-stack train services south of the City. Delays in executing this project are cited as the reason for the loss of these lines. The Port's North Container Terminal is presently dormant and the South Terminal is significantly under-utilized. As reported in WWS/World Wide Shipping, July/August 1994, Pg. 41: The Mayor of San Francisco announced a plan for a New Age entertainment center, incorporating a ballpark and a sports area to be built in space formerly used for cargo handling and Southern Pacific trackage—underscoring the trend to convert prime commercial waterfront land into resort and entertainment areas—an industry-wide problem (Containerization International, "San Franciscos Latest Setback," September 1994, p. 27).

Savannah, GA

Completion of a new 12.8 m (42 ft) shipping channel was completed this Spring. The 1.22 m (4 ft) deepening of the channel makes the Savannah terminals accessible to 98 percent of ships currently in the trade. 1994 is the third year of Savannah's $319 million development program called *Focus 222*, which is designed to provide the facilities and infrastructure needed to maintain growth into the year 2000. Remaining elements of the Program include steps to help restore the freshwater habitat in the Savannah National Wildlife Refuge, completion of upgrading the 1,680 m (5,500 ft) of contiguous berth at Garden City's Container Berth 6, the addition of 12 ha (30 acres) of container storage and delivery of four new post-Panamax container cranes, two of which were scheduled to arrive late in 1994, and upgrading of existing container cranes, making a total of 13 container cranes at the Garden City port complex (WWS/World Wide Shipping, May 1994, p. 27).

Seattle, WA

The ports of Seattle and Tacoma use the findings of a 1990 econometric study sponsored by the Washington Public Ports Association as an integral part of their planning strategies. In the case of Seattle, this means being capable of handling 2.1-2.5 million twenty-foot equivalent units annually, 15 years hence. The port's Container Terminal Development Plan, adopted by the Seattle Port Commission in May 1991, called for another 97 ha (240 acres) of land to be developed by the end of the century. A further 41 ha (100 acres) has been scheduled for possible acquisition by the year 2010. Seattle currently has about 140 ha (350 acres) of land that is dedicated to container handling activities. The initial phase of the Program involves adding parking space, extending certain piers and upgrading shipside cargo handling gear. Additionally, the Container Terminal Development Plan calls for expansion of existing, and construction of new on-dock rail yards, and improving overall access to/from the port area. A summary of Seattle's current expansion/development programs includes:

a. Expansion of Terminal 5, operated by an affiliate of American President Lines, from 33.6 has to 64 ha (83 to 158 acres) and a 122 m (400 ft) extension of the berth. Work is scheduled for completion in 2 to 3 years;

b. An on-dock intermodal rail facility at Terminal 5 capable of handling two full-length double-stack rail cars simultaneously plus capacity for storing two more, and an overpass to segregate rail and truck traffic;

c. A 36.4 ha (90 acre) expansion to the 44.5 ha (110 acre) Terminal 18 located on the eastern side of Harbour Island. The expansion will permit doubling of the existing intermodal on-dock rail yard from 28 to 56 double-stack rail cars. The new south intermodal rail yard will have separate rail access to avoid conflict with Terminal 5 rail traffic. Container aprons will be upgraded, and the terminal's seven container cranes will be upgraded, and/or replaced by post-Panamax capacity gantry cranes. Additional plans call for an addition of 4 ha (10 acres) to the northernmost extremity of the Terminal, increasing its size to 18.2 ha (45 acres) and the lengthening of the ship berth by 122 m (400 ft). Terminal 18 is the Port's largest common-user facility, and will be able to handle two post-Panamax vessels at the same time (Containerization International, July 1994, pages 87-90).

Tacoma, WA

Tacoma's 20-year, $450 million 2010 Blair Waterway terminal expansion program is equally ambitious, but its implementation will be geared to customer demand. Major elements of the 2010 Blair Waterway program, which is designed to enable the waterway to handle the largest containerships afloat include:

a. The addition of approximately 125 ha (309 acres) of new container terminal area, 11 berths, and 30 ha (75 acres) of new intermodal rail facilities at the Port;

b. Dredging of the main access channel to a depth of 13.7 m (45 ft), and construction of a new city bypass road with subsequent dismantling of the Blair Road Bridge. The bridge is slated to be removed by the end of 1995 and the entire West Blair terminal project is to be completed by the end of 1996;

Additional planned port improvements include the construction of two new container terminals on the north side of the Blair waterway and the new terminals have two berths and 20.2 ha (50 acres) of land. The second new terminal will be built at the existing Terminal 7 and will consist of a one-berth 20.2 ha (50 acre) facility. Spoil from dredging work is being used to fill in the Milwaukee Channel and increase the Sea Land terminal by 9.7 ha (24 acres). According to the econometric study cited above, Tacoma will need to be able to handle between 2.5 and 2.8 million twenty-foot equivalent units in the year 2010 (Containerization International, July 1994, pages 87-90).

Wilmington, DE

No immediate improvements identified.

Wilmington, NC

Long term development plans by the North Carolina State Ports Authority include studies for the deepening of the outer bar channel to 14 m (46 ft), the river and harbor channel to 13.4 m (44 ft), and development of a new marine terminal upstream of the existing port complex. Dredging was expected to begin in early summer 1994 and site development work for the new terminal is slated for fiscal year 1996 provided funding is available. Similar planning for a new marine terminal on Radio Island, adjacent to existing port facilities at Morehead City, is underway. The recently completed channel and harbor dredging to 13.7 m (45 ft) makes Morehead City one of the deepest ports on the East Coast (WWS/World Wide Shipping, May 1994, p. 26).

Attachment D2
Port Population Growth Factors (1990 - 2010)

U.S. Ports	Counties	1990	2010	Growth Factor
		East Coast		
Boston, Massachusetts	Suffolk	663,906	792,200	
	Norfolk	616,087	631,300	
		1,279,993	1,423,500	1.11
Elizabeth, New Jersey	Essex	778,206	757,200	
	Kings, NY	2,369,966	2,364,992	
	Hudson	553,099	566,600	
	Richmond, NY	385,224	463,529	
	Union	493,819	502,300	
		4,580,314	4,654,621	1.02
Philadelphia, Pennsylvania	Philadelphia	1,585,577	1,513,674	
	Camden	502,824	550,500	
	Gloucester	230,082	269,300	
		2,318,483	2,333,474	1.01
Eddystone, Pennsylvania	Delaware	547,651	508,557	
	Philadelphia	1,585,577	1,434,694	
		2,133,228	1,943,251	0.91
Wilmington, Delaware	New Castle	441,946	513,750	
		441,946	513,750	1.16
Baltimore, Maryland	Baltimore	692,134	728,898	
	Anne Arundel	427,239	499,204	
	Howard	187,328	288,701	
		1,306,701	1,516,803	1.16
Newport News, Virginia	Isle of Wight	25,053	34,283	
	Norfolk City	261,229	253,809	
	Hampton City	133,793	146,648	
	York	42,422	56,000	
		462,497	490,740	1.06
Norfolk, Virginia	Isle of Wight	25,053	34,283	
	Norfolk City	261,229	253,809	
	Portsmouth City	103,907	101,965	
	Hampton City	133,793	146,648	
	York	42,422	56,000	
		566,404	592,705	1.05
Portsmouth, Virginia	Isle of Wight	25,053	34,283	
	Portsmouth City	103,907	101,965	
	Norfolk City	261,229	253,809	
		390,189	390,057	1.00
Wilmington, North Carolina	New Hanover	120,284	150,936	
	Brunswick	50,985	79,644	
		171,269	230,580	1.35
Charleston, South Carolina	Charleston	295,039	339,400	
	Berkeley	128,776	252,800	
		423,815	592,200	1.40
Savannah, Georgia	Chatham	216,935	273,391	
	Byran	15,438	23,610	
		232,373	297,001	1.28

U.S. Ports	Counties	1990	2010	Growth Factor
Fernandina Beach, Florida	Nassau	43,941	79,800	
		43,941	79,800	1.82
Jacksonville, Florida	Nassau	43,941	79,800	
	Duval	672,971	1,014,100	
		716,912	1,093,900	1.53
Port Everglades, Florida	Broward	1,255,488	1,980,900	
		1,255,488	1,980,900	1.58
Miami, Florida	Dade	1,937,094	2,809,700	
		1,937,094	2,809,700	1.45
		Gulf Coast		
Mobile, Alabama	Mobile	378,643	408,600	
	Baldwin	98,280	110,300	
		476,923	518,900	1.09
Gulfport, Mississippi	Harrison	165,365	175,291	
		165,365	175,291	1.06
Galveston, Texas	Galveston	217,399	245,820	
	Brazoria	191,707	249,644	
	Chambers	20,088	21,200	
		429,194	516,663	1.20
New Orleans, Louisiana	Jefferson	448,306	513,980	
	Orleans	496,938	514,740	
	St. Bernard	66,631	79,950	
	Plaquemines	25,575	29,820	
		1,037,450	1,138,490	1.10
		West Coast		
Seattle, Washington	King	1,507,319	1,833,133	
	Kitsap	189,731	261,970	
		1,697,050	2,095,103	1.23
Tacoma, Washington	Pierce	586,203	792,179	
		586,203	792,179	1.35
San Francisco, California	Marin	231,200	245,500	
	San Mateo	652,100	787,300	
	San Francisco	723,900	781,700	
		1,607,200	1,814,500	1.13
Concord Naval Weapons, California	Contra Costa	810,300	1,096,300	
	Solano	345,700	557,400	
		1,156,000	1,653,700	1.43
Oakland, California	Alameda	1,279,182	1,561,900	
	San Francisco	723,959	781,700	
		2,003,141	2,343,600	1.17
Los Angeles, California	Orange	2,424,100	3,104,100	
	Los Angeles	8,897,500	11,441,900	
		11,321,600	14,546,000	1.28
Long Beach, California	Orange	2,424,100	3,104,100	
	Los Angeles	8,897,500	11,441,900	
		11,321,600	14,546,000	1.28

+1990 Census taken from Rand McNally/The New Cosmopolitan World Atlas Census/Environmental Edition, 1992.

Alabama	Alabama Population Projections 1990-2015, Alabama State Data Center Center for Business and Economic Research, University of Alabama, Tuscaloosa, AL, January 1994.
California	Population Projections by Race/Ethnicity for California and its Counties, Report 93 P-1, Demographic Research Unit, Sacramento, CA, (916) 322-4651, April 1993.
Delaware	Census info and projection numbers through Evelyn Pearson, Delaware Development Office, Business Research Section, Dover, DE, Consortium Series, (302) 739-4271, June 30, 1994.
Florida	Projected from Florida Population Studies (by county) by Stanley K. Smith, Director, Bureau of Economic and Business Research, University of Florida, Volume 27/Number 2/Bulletin No. 108, February, 1994.
Georgia	Census info and projection numbers through Marty Sik, Governor's Office of Planning and Budget, Atlanta, GA, (404) 656-0911.
Louisiana	Census info and projections provided by Division of Administration, Baton Rouge, LA, Department of Budget, ATTN: ARL, (504) 342-7410.
Maryland	Department of State Planning, Office of State Planning Data, Office of Michael Lettre, , Baltimore, MD, (410) 225-4452, September 29, 1994.
Massachusetts	Ms. Alice Rarig, Massachusetts Inst. for Social & Econ Research (MISER), University of Mass., Amherst, MA, (413) 545-6660, September 30, 1994. **Calculations are only preliminary numbers. Final reports will be made available by end of October 1994.
Mississippi	Projections given by phone through the Office of Dr. Barbara Logue (EPA) on 9/29/94 from Center Policy Research & Planning, MS Institute of Higher Learning, Jackson, MS, (601) 982-6576, September 29, 1994.
New Jersey	Census info and projection numbers provided by Sen-Juan Wu, New Jersey Dept of Labor, Labor Market & Demographic Rsr, Trenton, NJ, (609) 292-0076.
New York	Census info and projections provided by New York State Bureau of Economic and Demographic Info, Albany, NY, (518) 474-6005.
North Carolina	Census info and projection numbers through Bill Tillman, Office of State Planning, Raleigh, NC, (919) 733-4131, Prepared April 1994.
Pennsylvania	Projections given by David Gordner, Bureau of Water Management, Department of Environmental Resources, Harrisburg, PA, (717) 772-4048, September 30, 1994.
South Carolina	Census info and projection numbers through Diana Tester, South Carolina Budget and Control Board, Office of Research & Statistical Services, Columbia, SC, (803) 734-3619, Published October 29, 1993.
Texas	Census info and projection numbers obtained through Texas State Data Center, Texas A & M University System, College Station, TX, 77843-2125, (409) 845-5115. Contact: Hazel Dolar.

Virginia	Projection given by Jeanne Brown, Center for Public Service University of Virginia, Charlottesville, VA, (804) 982-5580, September 28, 1994.
Washington	Census info and projections from Washington State County Population Projections, Office of Financial Management, Forecasting Division, Olympia, WA, January 31, 1992.

Attachment D3
Background Discussion of Alternative Analytical Models for Evaluation of Potentially Impacted Port Populations

In the Fall of 1993, the Department of Energy (DOE) began to collect and analyze information required for the list of port criteria included in the Notice of Intent (DOE, 1993) for this environmental impact statement (EIS). DOE recognized that there would be public concern associated with consideration of potential ports of entry for the foreign research reactor spent nuclear fuel. Therefore, DOE decided to develop a sound technical basis for the identification of potential ports of entry.

As a result (concurrently with the independent evolution of the Urgent Relief Environmental Assessment), a list of 28 potential commercial ports was established based on the recommendations of independent maritime consultants. The database included information in the following categories:

1. *Geophysical Factors,* harbor and channel water depths [a port would fail if it had less than 7 m (23 ft) of water, but receive the maximum score if it had more than the 12 m (40 ft) of water required for all but the largest cellular container vessels]; the nautical distance from the open ocean to the port [ports greater than 40 km (25 mi) from open ocean received no points, but were not disqualified from further consideration]; and navigational factors that might increase public risks (narrow, winding channels with currents or other factors seriously affecting safe navigation were given no points, but a weighting factor was applied to channels with good characteristics to account for the relatively greater importance of this factor for maritime safety).

2. *Port facilities,* which included the capabilities of cargo terminals for handling containerized foreign research reactor spent nuclear fuel, wharves and depths alongside, crane capacities, terminal access (truck and rail), terminal security, and the liner services available.

3. *Factors related to spent nuclear fuel handling and transport,* including past experience with spent nuclear fuel or other hazardous cargoes, whether there were local restrictions on the receipt of foreign research reactor spent nuclear fuel, emergency response capabilities, hazardous material handling training, locations of terminals relative to nearby populations with a doubleweighted score for ports that were remote from urban populations (e.g., heart of a city), 1990 census statistics for port city populations and population densities, environmental factors (whether the immediate port vicinity had sensitive populations of animals), and distance from the port to Savannah River Site and Idaho National Engineering Laboratory (at that time these were the preferred storage sites due to historical experience and facilities; the other three sites were added later as the result of the Programmatic Spent Fuel Draft EIS decision to consider them).

Using the database developed, a semi-quantitative analysis of the port criteria was prepared that summed the "score" assigned to each port attribute by the maritime experts, and the ports were ranked from best to the least acceptable (this list of ports is, for the most part, a subset of the set of over 40 ports that were subsequently analyzed in detail in Section D.2 of this appendix).

DOE determined that a semi-quantitative analysis of all ports for all of the noticed criteria was unacceptably subjective, especially concerning the assignment and weighting of the numerical scores. Furthermore, it did not differentiate well between ports, and when weighting factors were applied to better discriminate between criteria that were very important to safety versus those that were "desirable attributes," the methodology became very difficult to justify.

Attachment D4
Derivation of Ship Collision Damage Probabilities

Derivation of the accident severity category probabilities requires that a probability of damage to the transportation cask, given a collision between two vessels, be calculated. In Appendix D, this probability has been characterized by two values, P_{Impact} and P_{Crush}. The first is a probability that the cask is damaged due to impact forces associated with the collision. The second represents the probability that crush forces result in damage to the cask. This attachment describes how these probabilities were derived.

D4.1 Kinetic Energy

V.U. Minorsky developed a method for analyzing the collision of ships that provides a correlation between resistance to penetration and the energy absorbed in the collision (Minorsky, 1959). The absorbed energy was determined for actual collisions by assuming the impact was nearly transverse, the hydrodynamic forces due to water entrained by the hull of the struck ship could be treated as a virtual increase in mass, and the collision was perfectly inelastic. The resistance to penetration was quantified through a resistance factor, R_t, which was computed from accident and ship design information. He found, for higher energy incidents, that there is a linear correlation between R_t and the absorbed energy.

ORI Inc., in a draft report on accident severities associated with water transport of radioactive materials, extended Minorsky's method to develop correlations between penetration depth and the energy absorbed in ship collisions (ORI, 1981b). By considering empirical probability distributions for displacement of the striking ship, its speed, and the angle of impact, bounding case curves were developed for the probability of occurrence of force levels at selected penetration depths. The force value referred to is the collision force acting between the two ships.

Only a fraction the collision force would be seen by a spent fuel transportation cask on board the struck ship. ORI gave a qualitative discussion of this aspect of the collision, together with some limiting case values based on assumptions about stowage and the presence and type of other cargo.

The present analysis depends, to a large extent, on the Minorsky and ORI analyses. It does add an approximate treatment of accelerations experienced by the spent fuel package and includes the effects due to cargo in determining the maximum penetration depth in collision events. The dynamics of inelastic collisions are treated through conservation laws for momentum and energy. Following Minorsky, the transverse hydrodynamic forces on the hull of the struck ship are accounted for by a virtual increase in mass, hence kinetic energy. This is a conventional method used by naval architects, but has limitations when applied to collisions. M.J. Petersen pointed out that experiments and calculations by Motora et al., have shown that the added mass treatment is not always a good approximation (Petersen, 1982; Motora, 1971). Here we accept the limitations imposed by the added mass method, because a more rigorous treatment of the collision is not warranted due to other uncertainties in the analysis, particularly in the modeling of cargo effects.

It should be noted that the ORI/Minorsky method of calculating hull penetration probably does not take account of the massive keel structures in the struck transport ships. Therefore, they most likely significantly overestimate the probability of penetration further than one-fifth of the beam of the struck ship, since penetration to this distance would mean that the keel structures had been encountered. Note that historic experience (rule-of-thumb experience) indicates that few ship collisions lead to penetration more than one-fifth of the beam of the struck ship.

Parameters and Assumptions

The target ship in the following calculations is assumed to have a beam of 24.99 m (82 ft) and a displacement, 'm', of 25,310 metric tons (27,841 tons). The virtual mass, 'dm', due to hydrodynamic forces is 0.4 m = 10,120 metric tons (11,132 tons). Eight cases are considered for the displacement, 'm', of the striking ship: 5,600; 16,800; 28,000; 39,200; 50,400; 61,600; 72,800; and 84,000 metric tons (6,160; 18,480; 30,800; 43,120; 55,440; 67,760; 80,080; and 92,400 tons. The normal component of the striking speed at impact ranges from 1 to 10 meters per second (1.9 to 19 knots or 2.2 to 22 statute miles per hour).

A full distribution of sailing speeds (0-22 knots) was used in the penetration calculations even though speeds in port channels are likely to be no greater than 10-15 knots and speeds at dockside only a few knots (minimum required to maintain steerage). In addition, large ships (e.g., tankers) are likely to be pushed/towed by tugs near docks.

The models for energy absorption by the ship and its cargo follow the methods of ORI. The work, 'W', due to cargo compression is the product of the crush strength of the cargo, the cross sectional area of the blunted bow of the striking ship, and the difference between the penetration distance and the cargo closeup distance. ORI gave examples of this calculation, which are reproduced in the formula

$$W_{cargo} = 19.44 f \sigma \,(x - f(\,beam\,))$$

where f is the fraction of open space on the hold floor, σ is the crush strength of the cargo in MPa (mega pascals), 'x' is the penetration depth and beam is the width of the struck ship, both in meters. This formula follows ORI in assuming the vertical size of the damage zone is 7.62 m (25 ft), and one third of the blunted bow is the effective area.

Prior to the initiation of cargo compression, energy is absorbed solely by deformation of the ship structure; this effect is modeled using the Minorsky value of 32 'mj' (mega joules) for the energy to penetrate the hull, together with the semi-empirical curves in Figure 6.2 of the ORI report. Table D4-1 gives coefficients for a quadratic fit used to represent the ORI curves below 15 m (49.2 ft) penetration, while a second fit for greater penetration distances is given in Table D4-2.

Table D4-1 Quadratic Coefficients for Energy Absorbed Due to Ship Structures <15m

$W_{ship} = a + bx + cx^2$ (x <15m)

Metric ton	a (mj)	b (mj/m)	c (mj/m^2)
5,600	9.551	0.6836	0.0405
16,800	8.709	0.8118	0.2984
28,000	8.056	1.2030	0.4558
39,200	8.121	1.0850	0.5296
50,400	9.234	0.6555	0.6217
61,600	8.956	0.8639	0.6698
72,800	8.574	1.1790	0.6906
84,000	8.204	1.5290	0.7154

Table D4-2 Quadratic Coefficients for Energy Absorbed due to Ship Structures >15m

$$W_{ship} = a + bx + cx^2 \quad (x > 15m)$$

Metric ton	*a (mj)*	*b (mj/m)*	*c (mj/m²)*
5,600	58.13	-4.837	0.1919
16,800	14.72	-0.057	0.3337
28,000	15.76	-0.304	0.5179
39,200	64.87	-5.622	0.7306
50,400	189.5	-19.21	1.162
61,600	264.4	-29.02	1.531
72,800	303.8	-36.36	1.878
84,000	412.2	-50.50	2.393

Distribution of Ship Displacements, Speeds and Angles

Analysis of two years of shipping accident data allowed ORI to develop probability distributions for 'M' (mass of the striking ship), 'V' (transverse speed of the striking ship), and θ (angle of incidence), which are presented here in Table D4-3 through D4-5. The ORI tables originally contained eleven intervals for displacement of the striking ship. Four cargo loadings were examined in the analysis: no cargo, light cargo, medium cargo, and heavy cargo (light, medium, and heavy refer to the amount of cargo on board). For the present work, the two lowest intervals were combined as were the three highest, yielding eight intervals to match the eight ORI curves for 'W'. There were also 11 values of 'V' in the ORI tables, with speeds ranging up to 11.3 meters per second (21.5 knots or 24.9 statute miles per hour), and 9 values of the collision angle. Thus 968 different combinations of these values are treated in determining transportation cask failure.

Table D4-3 Probabilities for Striking Ship Displacement

Displacement (metric ton)	*Probability of Occurrence*
0 - 10,160	0.15
10,161 - 20,321	0.25
20,322 - 30,481	0.25
30,482 - 40,642	0.05
40,643 - 50,802	0.05
50,803 - 60,963	0.05
60,964 - 71,123	0.10
71,124 - 152,407+	0.10

Table D4-4 Probabilities of Striking Ship Speeds

Speed (meters/second)[a]	*Probability of Occurrence*
0.0 - 1.028	0.0448
1.028 - 2.058	0.2538
2.058 - 3.087	0.1045
3.087 - 4.115	0.1343
4.115 - 5.144	0.1343
5.144 - 6.173	0.0896
6.173 - 7.202	0.0746
7.202 - 8.231	0.0597
8.231 - 9.260	0.0746
9.260 - 10.29	0.0149
10.29 - 11.32	0.0149

[a] *1 meters per second = 1.9 knots = 2.2 miles per hour*

Table D4-5 Probabilities of Striking Ship Angles of Incidence

Angle From the Normal (degrees)	Probability of Occurrence
0 - 10	0.2754
10 - 20	0.1305
20 - 30	0.0725
30 - 40	0.1305
40 - 50	0.1015
50 - 60	0.0724
60 - 70	0.1303
70 - 80	0.0435
80 - 90	0.0434

Speed During a Collision

In the following, 'M' and 'V' are the mass and transverse speed of the striking ship, while 'm' and 'v' denote the mass and transverse speed of the struck ship. Theta (θ) is the angle of impact, measured from the normal to the direction of the struck ship (this is the angle used by ORI, Minorsky and Petersen use its complement). The amount of virtual mass attributed to the struck ship to account for transverse hydrodynamic forces is 'dm'. W(x) denotes the work done in deforming the ships and compressing the cargo during a penetration to a depth 'x', and E_0 is the initial kinetic energy in the motion of the striking ship transverse to the struck ship.

The total energy in the transverse motion of the striking ship is:

$$E = MV^2 \cos^2(\theta)/2$$

Because energy is conserved during the collision, and neglecting turning effects,

$$E = \frac{MV^2}{2} + \frac{(m+dm)\,v^2}{2} + W(x)$$

Because momentum is conserved,

$$MV\cos(\theta) = MV + (m+dm)\,v$$

Together these equations yield a quadratic expression of the velocity of the struck ship:

$$\frac{Av^2}{2} - V\cos(\theta)\,v + \frac{W(x)}{m+dm} = 0$$

where A = (1+(m+dm)/M).

The value of the struck ship's transverse speed during the collision is, therefore,

$$v = \frac{V\cos(\theta)}{A} - \frac{1}{A}\sqrt{V^2\cos^2(\theta) - \frac{2AW(x)}{m+dm}}$$

The second term in this equation decreases to zero during the collision, yielding a terminal speed of $V\cos(\theta)/A$. This is also the terminal speed component of the striking ship in the same direction. The change in kinetic energy is (1-1/A)E = (m+dm)/(M+m+dm)E, in agreement with Minorsky.

Maximum Penetration Distance

The maximum penetration of the bow of the striking ship into the target ship was computed by finding, using Newton's method, the position at which the ships reached their terminal speed. From the conservation laws for energy and momentum, the condition for this to occur is:

$$0.5\,\mu\,V^2\cos^2(\theta) = W(x)$$

where $\mu = M(m+dm)/(M+m+dm)$, and $V_{\cos(\theta)}$ is the initial normal speed of the striking ship. For the no cargo case, it was found that for each of the striking ship displacements considered, initial normal speeds of 8 meters per second (15.2 knots or 17.6 statute miles per hour) and 10 meters per second (19.0 knots or 22.0 statute miles per hour) were sufficient to cut completely through the struck ship, resulting in a probable sinking; refer to Figure D4-1. On the other hand, at 2 meters per second (3.8 knots or 4.4 statute miles per hour) only the four heavier ships would even penetrate the hull of the struck ship, and at or below 1 meters per second (1.9 knots or 2.2 statute miles per hour) the hull was not punctured for striking ships of any displacement.

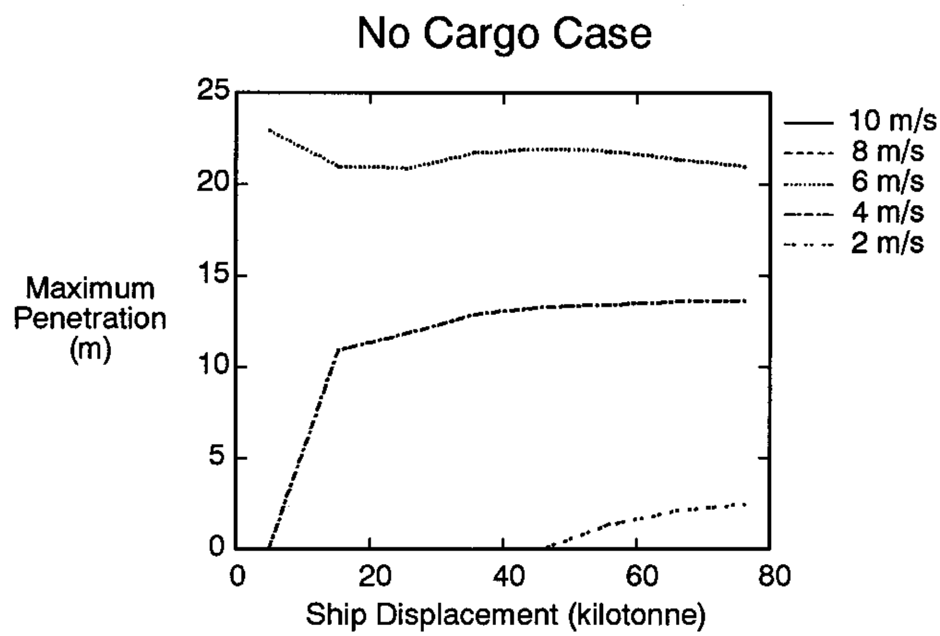


Figure D4-1 Maximum Penetration Distance in the No Cargo Case

Figure D4-2 shows the corresponding information for the light cargo case. Because of the packing fraction for this case, 0.6, the cargo effect does not begin until penetration has reached 15 m (49.2 ft). The figure shows the results as a function of the displacement of the striking ship, for normal impact speeds from 2 meters per second (3.8 knots or 4.4 statute miles per hour) to 10 meters per second (19.0 knots or 22.0 statute miles per hour). There were no cases where the struck ship would be completely cut through. At the two lower speeds, the cargo did not close up, hence was not a factor in absorbing the impact energy. There was no penetration at 1 meter per second (1.9 knots or 2.2 statute miles per hour) for any of the eight striking ship displacements considered.

The medium and heavy cargo results are shown in Figures D4-3 and D4-4, respectively. Figure D4-3 shows the cargo effect beginning at 5 m of penetration, and is important down to impact speeds of 4 meters per second (7.6 knots or 8.8 statute miles per hour). The cargo did not close up at smaller speeds, so was not a factor in determining the penetration depths. A similar result was obtained for heavy cargo; in both cases there was a strong influence by the cargo on the maximum penetration depth.

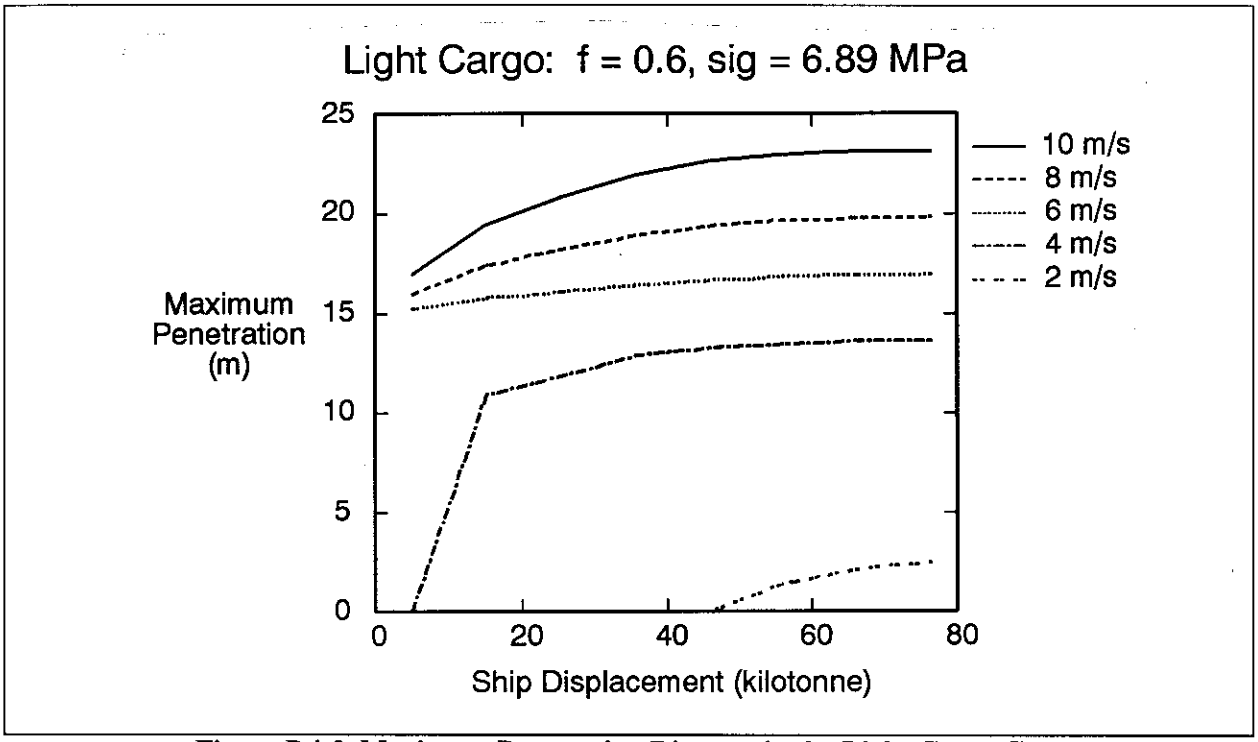


Figure D4-2 Maximum Penetration Distance in the Light Cargo Case

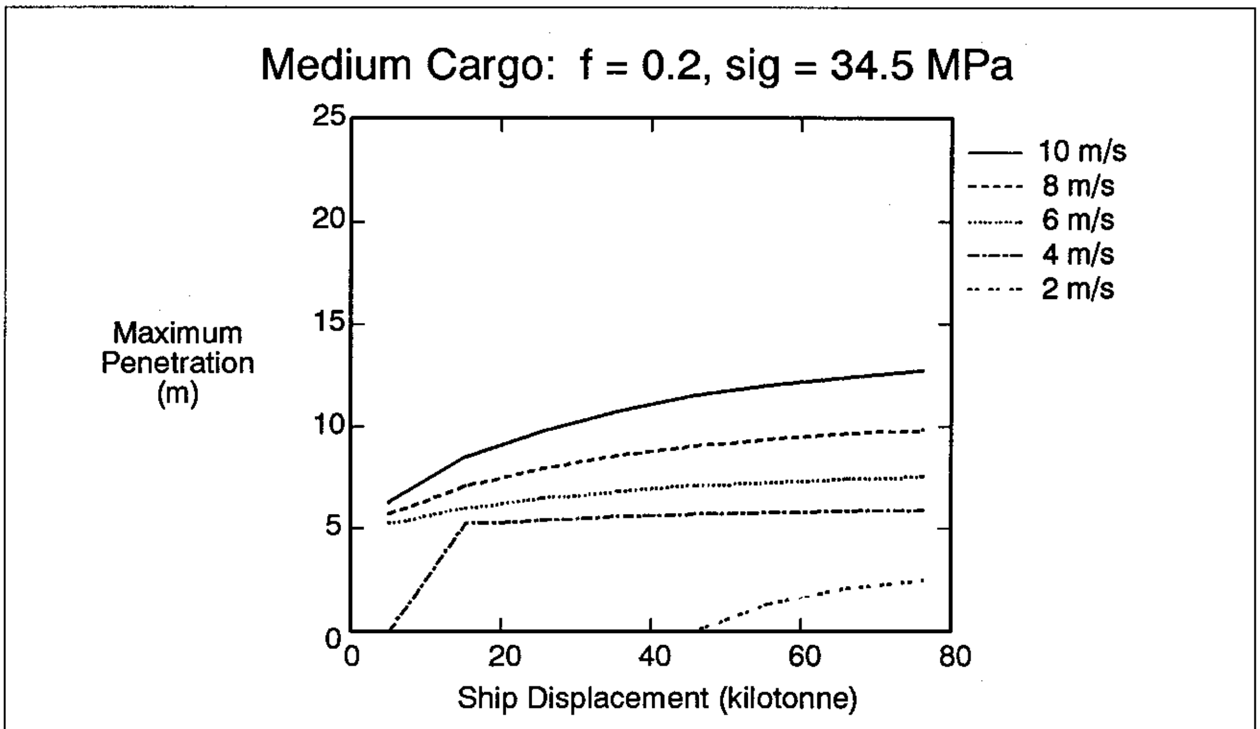


Figure D4-3 Maximum Penetration Distance in the Medium Cargo Case

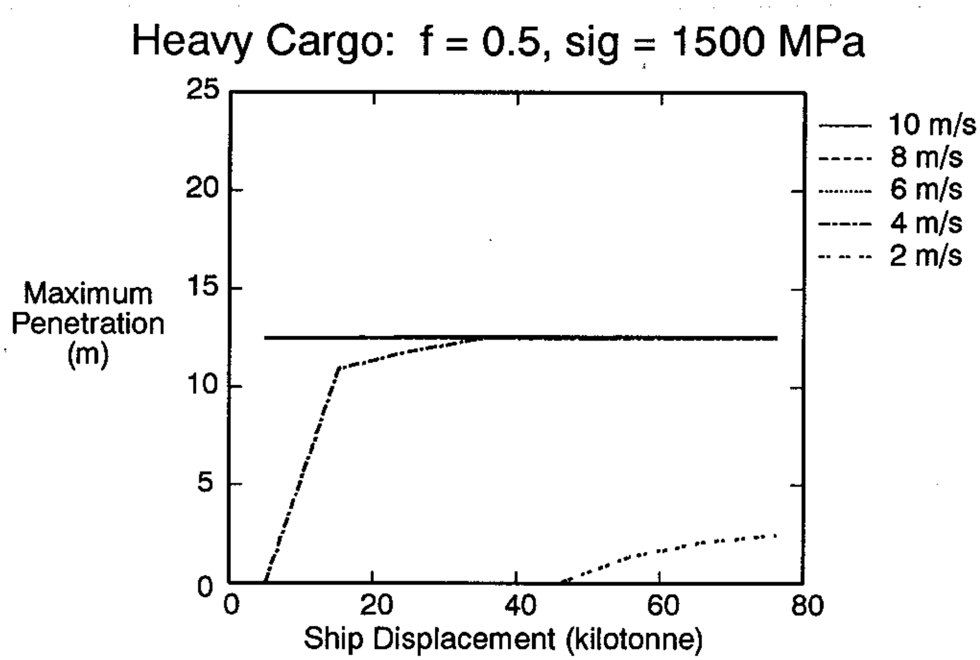


Figure D4-4 Maximum Penetration Distance in the Heavy Cargo Case

D4.2 Impact Forces During the Collision

Fuel elements experience impact forces if during a strong acceleration event they are driven against the inside of the cask or basket, or come into hard element to element contact. It is shown in Sanders et al., (Sanders, 1992) that accelerations of 100 g can be produced in the hypothetical accident conditions defined by NRC, which involve 9-m (29.5-ft) drops onto unyielding targets (NRC, 1990). They also showed there is a resulting cladding breach probability that for some power fuel types can be up to 0.0002 per rod in such events. We show here that the average acceleration experienced in ship collisions is very much smaller, usually below 1g, and conclude that inertial effects on the fuel are not significant for ship collisions.

The acceleration as a result of a ship collision is the time derivative of the transverse speed of the struck ship:

$$a = \frac{dv}{dt} = \frac{dv}{dx}\frac{dx}{dt} = v\frac{dv}{dx}$$

Performing the derivatives yields:

$$a = \frac{v(x)\,F(x)\,/(m + dm)}{\sqrt{V^2\cos^2(\theta) - \dfrac{2A\,W(x)}{m + dm}}}$$

where F(x) = dW(x)/dx. Notice that the acceleration peaks at the end of the collision, because the argument of the square root goes to zero there. The acceleration has a vertical asymptote at the maximum distance of penetration; the average acceleration, however, remains small.

The average acceleration during the collision is:

$$<a> = \frac{\int_o^d a(x)dx}{d} = \frac{V^2 \cos^2(\theta)}{2\, d\, A^2}$$

where d is the maximum penetration depth. This is an improper integral since a (x) has a singularity at 'd'. However, the integrand increases sufficiently slowly in the neighborhood of 'd', like $(x\text{-}d)^{-1/2}$, for the integral to converge.

When there is no other cargo in the hold with the spent fuel cask, the average acceleration is only a fraction of 1g (9.8 meters per second^2) in all cases, with the average acceleration always less than 5 meters per second^2. Similar results hold for the light and medium cargo cases. Even in the extreme case of heavy cargo, the average accelerations found were less than 2.5g. The highest acceleration, corresponding to a 75,000 metric tons (82,500 ton) ship striking with a normal speed of 10 meters per second (19.0 knots or 22.0 statute miles per hour), was about 0.2g (2 meters per second^2).

Because of these low average accelerations, generally on the order 1 percent relative to the accelerations expected in the NRC regulatory accident conditions, impact of fuel elements inside the cask is not expected to do any damage to the fuel as a result of collisions either in port or on the high seas. We conclude P_{impact} = 0.0.

D4.3 Crush Loads on the Fuel Package During the Collision

The spent fuel package of interest is the Pegase transportation cask, a cask of french design. It is a lead shielded cask, with a mass of 18.9 metric tons (20.8 ton), a diameter of 1.875 m (6.2 ft), and a height of 2.239 m (7.3 ft). It has a body composed of two stainless steel shells built around a lead shield. It is designed to carry fuel or other radioactive material in baskets of differing design which fit into the cylindrical cavity of the cask. A detailed analysis of the mechanical response of the Pegase transportation cask to crush forces is not available, however it is similar in construction to the lead shielded cask analyzed in the study of Fischer et al., (1987).

Fischer et al., developed a curve for the static force versus deflection for sidewise loading of a cask which was 4.9 m (16.1 ft) high, with a lead shield 0.133 m (0.4 ft) thick enclosed by an outer layer of stainless steel 0.0318 m 1.25 in) thick and an inner layer 0.0127 m (0.5 in) thick. Because a Pegase transportation cask is much shorter, but of similar construction, it will be at least as resistant to sidewise loading as Fisher's generic lead shielded cask. Fisher's results show that it requires a load of about 8.9 million newtons to produce a deflection of the cask body of 0.0254 m (1 in). A deflection of 0.0254 m (1 in) is judged to be a conservative deflection that could occur without damage to the fuel. Said another way, sidewise cask loading on a Pegase transportation cask in excess of 8.9 million newtons would probably result in some disruption of the fuel. Now the questions is, can crush forces on the cask as high as 8.9 million newtons be produced in a ship collision? To the extent that the homogeneous cargo models are applicable, the answer is "yes." The force applied by the cargo in these models, after closeup, is a constant equal to:

$$F_{cargo} = \sigma\, hd$$

where σ is the cargo crush strength, and 'hd' is the cross sectional area of the cask; for the Pegase transportation cask, 'hd' is 4.198 m^2 (45.2ft^2). Thus, the force in the light cargo case is 56.0 million newtons, and for the medium and heavy cargo cases it is many times larger. These values so far exceed the damage threshold at 8.9 million Newtons that major damage to the fuel and cask can be expected.

But if the cargo does not close up because the penetration is shallow or there is no other cargo in the hold, the cask does not see this force. Then, unless it is within the penetration region, it will not be significantly affected.

Inside the penetration region the cask can be crushed without the cargo going solid, or even if there is no other cargo in the struck hold. Cask tiedowns are designed, under U.S. regulatory practice, to withstand about 5 million newtons of transverse force (NRC, 1990). The difference between this value and the 8.9 million newtons required to produce a 0.025 m (1.0 in) deflection in the cask wall of the generic cask is not considered significant; moreover in ORI's opinion "the RAM [Radioactive Material] package could conceivably be restrained from sliding, even in an empty hold, after the fittings failed. A buckled deck for example could do this and in effect act as an infinitely strong fitting" (ORI, 1981a).

Thus there are two cases to consider for failure due to crush forces. In the first the penetration depth exceeds the cargo close-up distance, while in the second it exceeds the cask stowage location. We assume fuel damage and closure failure in both types of events.

Cask Failure Probability

This section evaluates the probability that a cask will fail when the ship carrying it is struck in a collision with another ship. Since there are two different scenarios, the total probability of cask failure is the sum of two terms, one of cargo going solid, the other for the ship over-running the cask location, or

$$P_{\text{crush}} = P_{\text{solid}} + P_{\text{contact}}.$$

P_{solid} and P_{contact} were evaluated by comparing the maximum penetration distance against the closeup distance and the stowage position, assumed to be on the centerline of the hull, for all combinations of striking ship displacement, speed, and angle given in Tables D4-3 to D4-5. Each individual case was counted as either resulting in cask failure (meaning the fuel is damaged and the cask seal is broken) or not, and the probability of the case was assigned according to the probability values in the referenced tables. The sum $P_{\text{solid}} + P_{\text{contact}}$ of the probabilities of all failure cases is P_{crush}.

The results are shown in Figure D4-5. The successive columns refer to the four models considered, for no cargo, and light, medium and heavy cargo. For other than the medium cargo model, the total crush probability is about 0.29, although the fraction due to the cargo going solid varies from 0 for the no cargo case to 1 for the heavy cargo case. The medium case, which as the smallest fraction of open hold space at 0.2, also has the highest failure rate, about 0.45. Of the four cases considered, this is the only case where the cargo goes solid well before the midline of the ship is reached, thus permitting a greater proportion of all the collisions to be significant from a cask damage point of view. Since this case shows the greatest probability, it is conservative to take P_{crush}= 0.45.

Alternate Case

Because the top speed in a harbor is controlled, the ORI distribution was adjusted to a top speed of 8.23 meters per second (15.6 knots or 18.1 statute miles per hour). This reduced the number of speed intervals to eight, and eliminated the three highest speed categories in Table D4-4. The total number of combinations of striking ship displacement, speed, and angle was therefore reduced from 968 to 704. Figure D4-6 shows the revised cask failure probabilities for the four cases. The highest failure probability is still from the medium cargo case, probably because this case has the earliest cargo closeup distance and fails most often from collisions which do not penetrate far into the target ship. The failure probability goes down more in the other cases because they involve penetrations going past the midline of the ship. Such events are sensitive to the high end of the speed distribution. The cask crush probability for this alternative is set equal to the largest result, $P_{\text{crush}} = 0.40$.

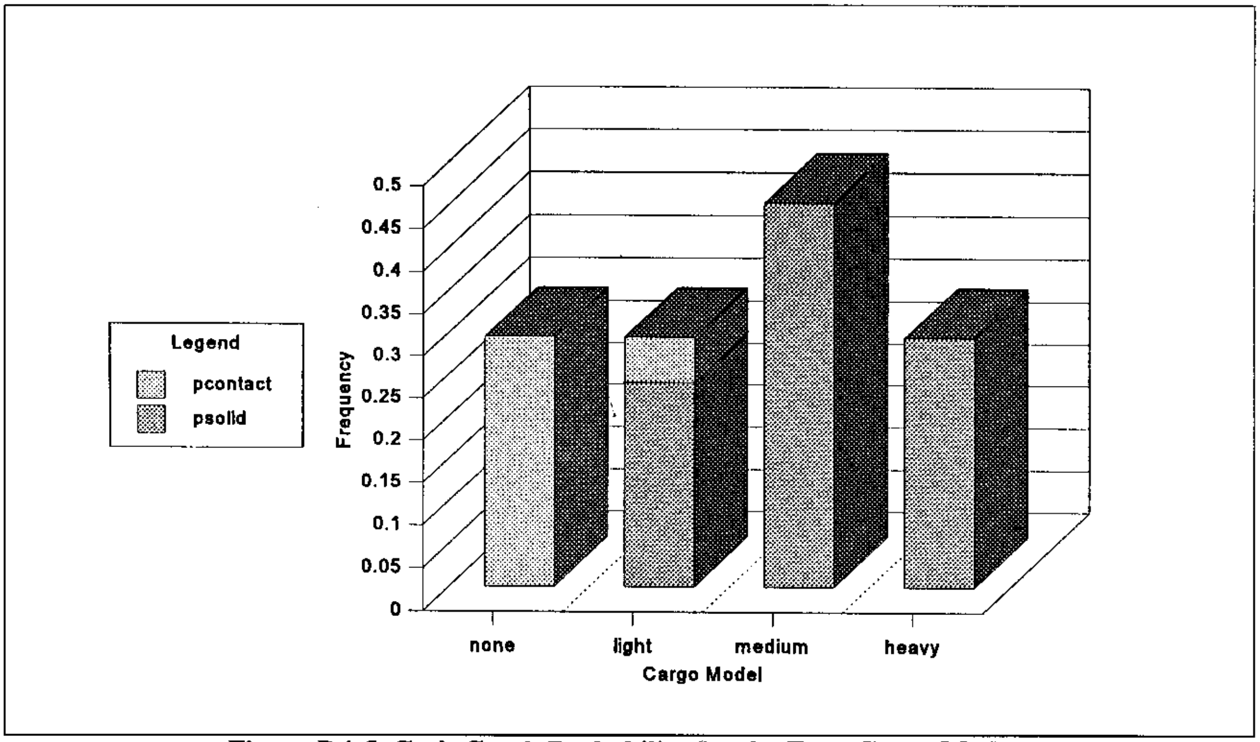


Figure D4-5 Cask Crush Probability for the Four Cargo Models

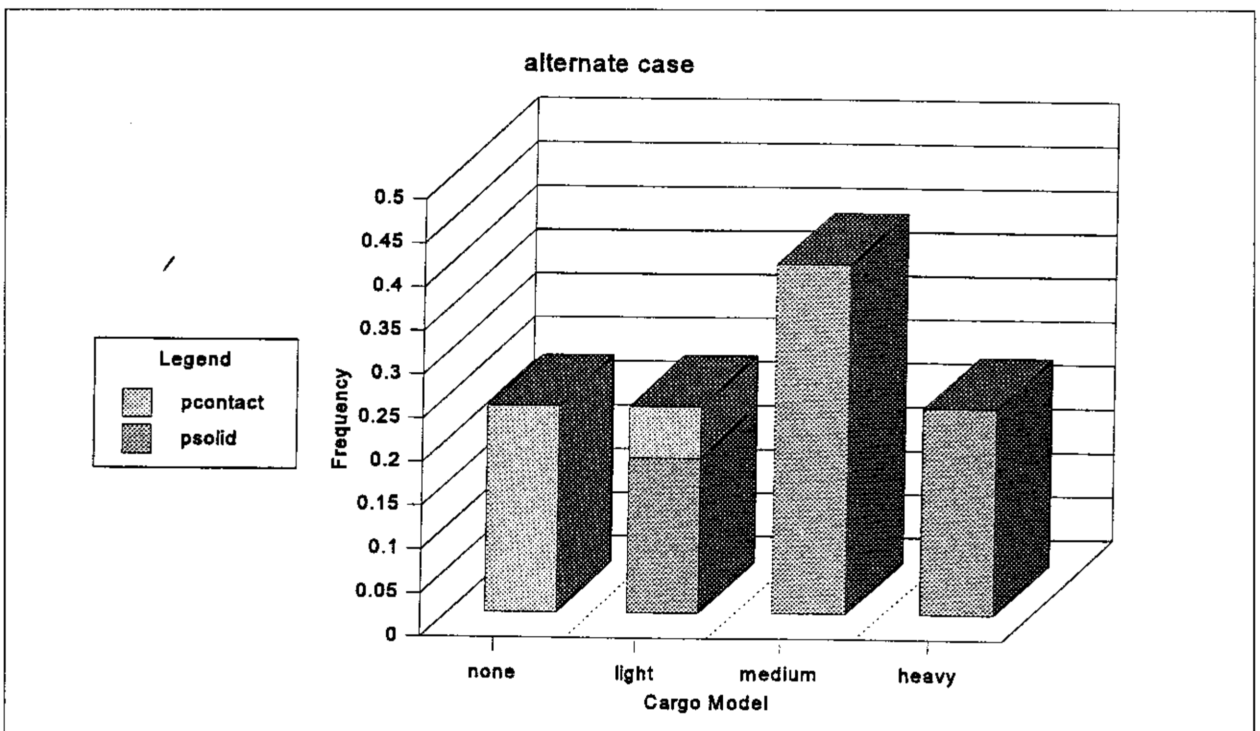


Figure D4-6 Cask Failure Probabilities for the Four Cargo Models with Striking Ship Speeds Truncated at 8.23 m/s

Attachment D5
High Temperature Effects on Research Reactor Fuel Release Fractions

D5.1 Introduction

Previous assessments of the accident risks associated with the transport of research reactor fuel did not specifically address certain high temperature (somewhat above 900°K or 1160°F) effects on the fuel. In this temperature range, aluminum based fuels (the aluminum-uranium alloys used in research reactor fuels) are susceptible to melting. Additionally, TRIGA fuel is spontaneously combustible in this same temperature range, if sufficient oxygen is available. The melting point for the uranium dioxide fuels that had been used as the basis for the development of estimates of the release fractions for earlier assessments is considerably higher than for the aluminum based fuels. These earlier assessments were the basis for the release fractions used in the base case analysis of this study. This attachment provides an assessment of the impact of these high temperature effects on the release fractions, and the probabilities of the accident severity categories, used in the base case study. This assessment forms the basis for the sensitivity study provided in Section 5.4.3.2 of Appendix D.

D5.2 Fission Product Release At High Temperatures

Table D-21 shows that accident severity category 4 accidents are caused by a ship collision that fails the seal of the spent fuel transport cask; that category 5 accidents add a severe engulfing fire to the conditions that characterize category 4 accidents; and that category 6 accidents assume an engulfing fire and a more severe cask failure (one medium sized hole or two or more small holes), one that allows the severe fire to induce substantial convective flow of air through the failed cask. Table D-21 also shows that the principal difference between severity category 5 and severity category 4 release fractions is a five order-of-magnitude increase (from 1.0×10^{-8} to 9.0×10^{-4}) in the release fraction for cesium; and that the principal difference between severity category 6 and severity category 5 release fractions is a 42-fold increase (from 1.0×10^{-6} to 4.2×10^{-5}) in the release fraction for ruthenium.

Much increased cesium volatility at the elevated temperatures to which the spent fuel is heated by the severe fire is the cause of the five order-of-magnitude increase in cesium release assumed for category 5 accidents. Conversion of elemental ruthenium to volatile ruthenium oxide (RuO_4) by oxygen, due to convective air flow through the failed cask, is the cause of the 40 fold increase of ruthenium release assumed for category 6 accidents.

The release fractions listed in Table D-21 and used in the base case analysis were constructed from release estimates developed (Wilmot, 1981; Wilmot et al., 1981) for power reactor fuel (uranium dioxide pellets clad in zircaloy). The fuels used in research reactors are not uranium dioxide pellets clad in zircaloy. TRIGA reactors use large pellets formed from a mixture of uranium, zirconium, and zirconium hydride (ZrH_2) that are clad in stainless steel. All of the other reactors considered in this assessment (BR-2 and RHF) use aluminum-clad metallic fuels where the metal is an alloy of aluminum and uranium (Al-U). At elevated temperatures (above 900°K or 1,160°F) these fuels melt and, if exposed to air, TRIGA fuel burns. Therefore, if a ship collision leads to a fire that heats these fuels to temperatures much above 900°K (1,160°F), fission product releases from these fuels will differ markedly from that predicted for uranium dioxide power reactor fuels. Therefore, the properties of metallic aluminum-uranium alloy fuels and of TRIGA fuel were reviewed to identify any significant differences between releases from these fuels and releases from power reactor fuel when these fuels are heated to elevated temperatures.

The BR-2, and RHF fuels considered by this study are fabricated as stacks of aluminum-uranium alloy plates or cylinders that are contained in aluminum-cladding. Release of fission products from aluminum-uranium alloy fuels has been reviewed (Ellison, 1993). Fission product release is minor below about 923°K (1,202°F), the melting point of the aluminum-uranium alloy from which these fuels are fabricated. Once the aluminum-uranium alloy has melted, fission products volatile at melt temperatures are rapidly released to the gas space above the molten alloy. Although molten aluminum can dissolve both the stainless steel spacers that support individual fuel bundles and the alloy fuel plates, melting of the aluminum-cladding that surrounds these alloy fuels does not significantly affect release because the melting temperature of the clad, 933°K (1,220°F), is slightly higher than the melting temperature of the alloy fuel.

The effects of air ingression on the release of fission products from commercial reactor fuel have been reviewed (Powers, 1994). That review indicates that ruthenium release fractions from uranium dioxide fuel will equal or exceed $4.2x10^{-5}$, the release fraction for ruthenium used in the base case analysis for category 6 accidents, if the fuel is exposed to air for 15 to 30 minutes while heated to 700°K (800°F). The review also indicates that release increases rapidly as temperature rises or exposure times lengthen, and that for temperatures less than 1,200°K (1,700°F), ruthenium is released principally as ruthenium-oxide.

TRIGA fuel is a uranium-zirconium-hydrogen alloy that burns spontaneously in air at temperatures above 925°K or 1,205°F (Benedict, 1981). Because this combustion process is highly exothermic, if a severe fire heats a failed cask containing TRIGA spent nuclear fuel to temperatures above 925°K (1,205°F), air ingression due to convection or contraction of cask gases upon cask cooling would be expected to initiate spontaneous combustion of the fuel alloy, which should lead to substantially increased release from the fuel to the cask interior of krypton, cesium (most likely as cesium hydroxide, CsOH), and ruthenium [by conversion to volatile ruthenium oxide (RuO_4)].

Theoretical (NRC, 1988; SNL, 1989; GNS, 1993; Shaffer, 1994) and experimental (Babrauskas, 1986b; Nelsen, 1986; Gregory, 1987; Gregory, 1989; Schneider, 1989; Keltner, 1994) estimates of the thermal loads on casks produced by engulfing fires indicate that only engulfing fires with durations of an hour or more caused by the combustion of high-grade fuels (gasoline, jet fuel, diesel fuel) with an ample oxygen supply can raise the spent fuel contained in the casks to temperatures that approach 1,000°K (1,340°F). These studies also indicate that cask temperatures this high are not attained for fires of similar duration caused by poorer fuels (e.g., crude oil, wood). Thus, short duration fires involving low-grade fuels or mixtures of low and high-grade fuels are unlikely to raise cask temperatures high enough to significantly increase cesium vaporization or to cause substantial conversion of ruthenium to volatile ruthenium oxide. Fires involving high-grade fuels that are oxygen-starved because hold covers are closed or suppressed by the operation of fire fighting systems are also unlikely to result in elevated release fractions. Conversely, engulfing fires of about one hour duration that involve high-grade fuels could, for some accidents, be able to heat cask interiors to temperatures where (1) aluminum-uranium alloy fuels melt, (2) krypton, cesium, and ruthenium are easily vaporized, both from TRIGA fuel pellets and from melted aluminum-uranium alloy fuels, and (3) conversion of ruthenium to ruthenium oxide is substantial, if either fuel is exposed to air. The impact of these high temperature affects on the accident severity category 5 and 6 release fractions are discussed in the following paragraphs.

Accident Severity Category 5

When fuel temperatures remain below 900°K (1,160°F), that is, below the ignition point of TRIGA fuel in air and the melting point of aluminum-uranium alloy fuels, the release fractions from TRIGA fuel should be similar to that from uranium dioxide fuels. Also, the releases from aluminum-uranium alloy fuels should be very small, perhaps negligible, since diffusion in the metal plates from which the fuel is fabricated will be too slow to cause significant release to the cask, much less to the environment.

When research reactor fuels are heated significantly above 900°K (1,160°F) the release to the cask from TRIGA fuel pellets and from melted aluminum-uranium alloy fuels of krypton, volatile cesium, and ruthenium should be substantial (Cubicoitti, 1984; Cordfunke, 1990). Once released to the cask interior, transport of these fission products from the cask to the environment (past the failed cask seal) will only be efficient when the gases in the cask expand significantly due to heating of the cask to temperatures well above 900°K (1,160°F). For example, if melting of an aluminum-uranium alloy fuel at 923°K (1,202°F) causes essentially all of the krypton trapped in the fuel to be released to the cask interior, then further heating of cask gases to 1,023°K (1,382°F) by the fire will cause approximately 10 percent of the gases in the cask, including the krypton that escaped from the fuel to the cask interior, to be lost to the environment by expansion past the failed cask seal.

After the fire dies out, cooling of the hot cask will cause air to be drawn into the cask as the gases in the cask cool and contract. Thus, almost any hot fire of substantial duration will lead to substantial air ingression into a failed cask. Enhanced ruthenium release will then occur only if large amounts of fuel are exposed to the air, if this exposure occurs when the fuel is still hot enough to allow ruthenium to be oxidized to a volatile species, and if there is a transport process operating that causes the volatile ruthenium species to be released from the failed cask.

Because aluminum-uranium alloy fuels are molten at temperatures above 923°K (1,202°F), after air is drawn into the cask by cooling, if still molten, substantial exposure of fuel to air will occur, and therefore oxidation of ruthenium to ruthenium oxide will occur. However, after release to the cask interior, release of ruthenium to the environment can only occur by an inefficient transport mechanism, diffusion against the inflow of air since the cask is now cooling down. Thus, category 5 accident conditions, even those that reach unusually high temperatures, are not expected to significantly increase ruthenium release from aluminum-uranium alloy fuels, unless after dying down and drawing air into the cask, the fire flares up anew and again heats the cask to elevated temperatures whereupon gas expansion would transport some of the oxidized ruthenium vapors from the cask to the environment.

Because TRIGA fuel burns spontaneously and exothermically at temperatures above 900°K (1,160°F), if cask cooling draws air into a cask that contains TRIGA fuel while the fuel is still at such elevated temperatures, fuel burning will convert ruthenium to ruthenium oxide, and heating of the fuel and the cask gases by the highly exothermic oxidation of the hydride fuel will cause the oxidized ruthenium to vaporize, the cask gases to expand, and some of the vapors to be transported from the cask to the environment.

Accident Severity Category 6

During category 6 accidents, release from fuel to the cask interior of krypton, cesium, and ruthenium (after conversion to ruthenium oxide by exposure to air), occurs by the same processes that were just discussed for category 5 accidents. Gas convection through the failed cask is, by definition, substantial during category 6 accidents. Exposure of hot fuel to air causes substantial conversion of ruthenium to ruthenium oxide. Additionally, all vapors released from the fuel to the cask are transported from the cask to the environment by the convective flow of gases.

D5.3 Release Fractions for High-Temperature Events

The discussion presented in Section D5.2 indicates that, at elevated temperatures, release fractions for aluminum-uranium alloy and TRIGA fuels will differ substantially from those assumed in earlier studies of research reactor fuel transportation accidents for category 6 events and also for category 5 events that reach unusually high temperatures. To allow the consequences of such high-temperature events to be examined, the severity category strategy used in the base case analysis was modified by dividing both categories 5 and 6 into a low temperature and a high temperature category. Release fractions were then estimated for all of the categories in the modified strategy (categories 4, 5A and 5B, and 6A and 6B) and sensitivity calculations were performed to estimate the effects of the new release fractions on accident consequences.

Fire events that do not heat cask contents above 900°K (1,160°F) are placed in categories 5A and 6A. Fire events that heat cask contents above 900°K (1,160°F) are placed in categories 5B and 6B. Events that lead to seal failure are placed in category 4 and 5. Events that lead to cask failures (one medium hole, two or more small holes) that allow significant convective flow of gases through the failed cask are placed in category 6. Thus, transport of fission products released from fuel to the cask interior for category 5 events must be driven by expansion of cask gases due to heating of the cask by the fire, while for category 6 events, transport from the cask to the environments is efficiently driven by convective flow of gases through the cask. Table D5-1 summarizes the attributes of the modified severity categories.

Table D5-1 Category Attributes for the Modified Release Category Strategy

Category	Cask Failure Mode	Transport from Cask	Temperature of Cask Contents
5A	Seal Failure	Gas Expansion	T < 900°K
5B	Seal Failure	Gas Expansion	T > 900°K
6A	One medium hole, two small holes	Convection	T < 900°K
6B	One medium hole, two small holes	Convection	T > 900°K

Table D5-2 presents the release fractions developed for this modified strategy. Summarized in the footnotes of Table D5-2 are the basis for these release fractions. This table also compares the revised release fractions to the release fractions that were used in all of the base case calculations performed in this study. The sensitivity calculations that were performed using these new release fraction are described in Appendix D Section 5.4.3.2.

Table D5-2 Modified Release Fractions for Severity Categories 4, 5, and 6

Severity Category	Study	Fuel	Chemical Element Group			
			Krypton	Cesium	Ruthenium	Particulate
4	Base Case	Both	0.01	1.0x10-8	$1.0x10^{-8}$	$1.0x10^{-8}$
	Sensitivity Case	TRIGA	0.1	1.0x10-7	$1.0x10^{-7}$	$1.0x10^{-7}$
		Aluminum-uranium	$1.0x10^{-8}$	$1.0x10^{-8}$	$1.0x10^{-8}$	$1.0x10^{-8}$
5	Base Case	Both	0.1	$9.0x10^{-4}$	$1.0x10^{-6}$	$5.0x10^{-8}$
5A	Sensitivity Case	TRIGA	0.26	$1.0x10^{-3}$	$2.3x10^{-6}$	$1.3x10^{-6}$
		Aluminum-uranium	$1.3x10^{-7}$	$1.3x10^{-7}$	$1.3x10^{-7}$	$1.3x10^{-7}$
5B	Sensitivity Case	TRIGA	0.31	$1.1x10^{-2}$	$9.8x10^{-3}$	$3.3x10^{-4}$
		Aluminum-uranium	0.098	$9.8x10^{-3}$	$1.7x10^{-6}$	$3.0x10^{-7}$
6	Base Case	Both	0.11	$9.8x10^{-4}$	$4.2x10^{-5}$	$5.0x10^{-8}$
6A	Sensitivity Case	TRIGA	0.35	$1.6x10^{-3}$	$3.6x10^{-6}$	$2.0x10^{-6}$
		Aluminum-uranium	2.0x10-7	$2.0x10^{-7}$	$2.0x10^{-7}$	$2.0x10^{-7}$
6B	Sensitivity Case	TRIGA	1.0	0.3	0.3	0.01
		Aluminum-uranium	1.0	0.1	$1.6x10^{-5}$	$1.6x10^{-6}$

In order to develop release fraction values for the sensitivity study accident categories, several parameters need to be defined. These parameters are defined in Table D5-3.

Table D5-3 Definitions of Parameters used in the Sensitivity Study Accident Categories

F_{B1}	Fraction of fuel elements failed by the ships collision
F_{C1}	Release fraction for fission products from the fuel to the cask cavity due to the mechanical effects of the ship collision
F_{CE1}	Fraction of the fission products released to the cask cavity that escape from the cask in the absence of a fire
F_{FC2}	Fraction of fission products released from the fuel to cask cavity due to heating of the fuel from ambient temperature (T_a)to some elevated temperatures (T_f) less than 900°K
F_{B2}	Fraction of the fuel elements failed by burst rupture due to heating from T_a to T_f
F_{CE2}	1 - (T_a/T_f) where $T_a/T_f = V_a/V_f$ = the fraction of the gases in the cask at ambient temperature that remain in the cask after heating to Tf
F_{FC3}	Fraction of fission products released from the fuel to the cask cavity after the fuel has been heated to T_{FC3}(=temperature where aluminum-uranium fuel melts and TRIGA fuel burns if exposed to air
F_{B3}	The fraction of fuel elements failed by burst rupture due to heating from T_{FC3} to T_f
F_{CE3}	1 - (T_{FC3}/T_f) where $T_{FC3}/T_f = V_{TC3}/V_f$ = the fraction of the gases in the cask after heating to T_{FC3} that remain in the cask after further heating to T_f

Then, the release fraction (F_{R4}) for Category 4 events is given by

$$F_{R4} = F_{B1}F_{FC1}F_{CE1} \tag{1}$$

If the collision leads to a fire that heats the cask to elevated temperatures that do not exceed 900°K (1160°F) heating of the fuel may cause more fission products to be released from the fuel to the 900° cask cavity, and expansion of cask gases due to heating by the fire will cause a substantial fraction of the gas borne fission products to be transported from the cask interior through the failed cask seal to the environment. Thus, the release fraction (F_{R5A}) for Category 5A events is given by

$$F_{R5A} = F_{R4} + F_{B1}F_{FC1}(1 - F_{CE1})F_{CE2} + F_{B2}F_{FC2}F_{CE2} \tag{2}$$

If the collision has led to cask failures (a single medium hole or two smaller holes) that allow substantial convective flow through the cask, then all fission products released to the cask interior will be transported from the cask to the environment. Thus, the release fraction (F_{R6A}) for Category 6A events is given by

$$F_{R6A} = F_{R4} + F_{B1}F_{FC1}(1 - F_{CE1}) + F_{B2}F_{FC2} \tag{3}$$

as by definition $F_{CE2} = 1.0$ for Category 6 events.

The release fraction (F_{R58}) for fire events that heat the cask to temperatures above 900°K (1160°F), i.e. Category 5B events where Al-U alloy fuels melt and TRIGA fuel burns if exposed to oxygen is given by

$$F_{R5B} = F_{R5A} + F_{B3}F_{FC3}F_{CE3} \tag{4}$$

where

Again, if a Category 6 event has occurred, the release fraction (F_{R6B}) will be

$$F_{R6B} = F_{R6A} + F_{B3}F_{FC3} \tag{5}$$

since by definition $F_{CE3} = 1.0$ for Category 6 events.

The release fractions used in the base case assessment are the same as those (Wilmot 1981) developed for air-cooled casks for release of fission products from spent commercial UO_2 fuel for three processes: impact, burst, and oxidation. Base case Category 4 release fractions are the same as those developed by Wilmot for impact events involving air-cooled casks. Except for cesium, Category 5 release fractions are equal to the sum of Wilmot's release fractions for impact and burst, and Category 6 release fractions are equal to the sum of Wilmot's release fractions for impact, burst, and oxidation. For cesium, the base case uses release fractions that have been adjusted somewhat to reflect the effect of metallic fuel properties on cesium release. This information is used as the basis to derive several of the values for the parameters identified in Table D5-3.

For impact events, Wilmot uses $F_{B1} = 0.1$, $F_{FC1} = 0.2$ and $F_{CE1} = 0.5$ for krypton; and $F_{FC1} = 2x10^{-6}$ and $F_{CE1} = 0.05$ for cesium, ruthenium and particulates for release of fuel fines and thus the fission products trapped in the fines. For burst events, Wilmot assumes that $F_{B2} = 0.9$. Table D5-2 shows that the base case used values of 0.1, $9x10^{-4}$, $1x10^{-6}$, and $5x10^{-8}$, respectively, for the release fractions for krypton, cesuim, ruthenium, and particulates for Category 5 events. If Equation 2 is solved for F_{FC2} using the base case values for Category 5 events for F_{R5A} and Wilmot's values for F_{B1}, F_{B2}, F_{FC1}, and F_{CE1}, then the following values are obtained for F_{CE2}: 0.15 for krypton, $1.6x10^{-3}$ for cesium, $1.6x10^{-6}$ for ruthenium, and 0 for particulates.

The analysis presented in Attachment D4 of cask damage caused by impact and crush concludes that damage will not result from the impacts forces experienced by cask during ship collisions, and that if the cask is subjected to crush forces, they will always be large enough to fail all of the fuel elements contained in the cask. Therefore, $F_{B1} = F_{B2} = F_{B3} = 1.0$.

To facilitate comparison of the new release fractions developed here to the release fractions used in the base case, the release fractions for the cesium, ruthenium, and particulate chemical element groups for Category 4 events were forced to be the same as the value used in the base case. Although aluminum-uranium alloy fuels should have very little, if any, fuel fines associated with the metal plates from which the fuel bundles are fabricated, to achieve this equivalence, it was assumed that aluminum-uranium alloy fuels have amounts of fuel fines one-tenth of those assumed by Wilmot for uranium dioxide fuels. Thus, for aluminum-uranium alloy fuels, $F_{FC1} = 2x10^{-7}$ and therefore, because $F_{B1} = 1.0$, $F_{R4} = 2x10^{-8}$, which is the value that the base case used for the realease fraction for cesium, ruthenium, and particulate for Category 4 events.

Reasonable choices for F_{FC3} for aluminum-uranium alloy fuels, that is, for release to the cask cavity upon melting for the alloy fuel, are 1.0 for krypton, 0.1 for cesium, $1.6x10^{-5}$ for ruthenium, and $1.6x10^{-6}$ for particulate, where ruthenium release from metallic fuel upon melting has been assumed to be ten times the ruthenium release from commercial uranium dioxide fuel estimated for Category 5A events (the value of F_{FC2} for ruthenium release from uranium dioxide fuel for Category 5A events. Particulate release has been assumed to be about the same as ruthenium release from uranium dioxide fuel for Category 5A events and about ten times larger than particulate releases from aluminum-uranium alloy fuels for Category 4 events (the value of F_{FC1} for particulate release from aluminum-uranium alloy fuels for Category 4 events), as the melting of aluminum uranium alloy fuels due to heating of the cask by a fire is not likely to be violent.

Reasonable choices for F_{FC3} for TRIGA fuel are 1.0 for krypton; 0.3 for cesium; 0.3 for ruthenium, since burning of the fuel means that ruthenium will be converted to a volatile oxide by exposure to air; and 0.01 for particulate, on the assumption that the high exothermicity of the combustion process will cause one percent of the fuel mass to be aerosolized. For Category 5B, these values were decreased by a factor of 3, because air can only enter the cask due to cooling, which will not lead to fuel burning if the fuel cools below 900°K, (1160°F). Even if burning does occur, efficient transport of fission products released by the burning from the cask to the environment can occur only by gas expansion caused by the heat released by fuel burning. Thus, the cask atmosphere must breath (pass through several cooling/burning cycles), if significant quantities of fission products are to be released by fuel burning, when there is no convective flow of air through the cask.

Table D5-4 lists the parameters used in Equations 1 through 5, and presents the values used for each parameter to calculate values for the release fractions F_{R4}, F_{R5A}, F_{R6A}, F_{R5B}, and F_{R6B}. For the four EA5 results for UO_2 fuel, the result calculated is the F_{FC2} value, not the F_{R5} value, which is an input and is set equal to the value used in the EA for the indicated element group.

Table D5-4 Parameters Used to Generate High Temperatures Fire Sensitivity Study Release Fractions

Accident Category	*Fuel*	*Element*	*Parameter*[1]									
			F_{B1}	F_{C1}	F_{CE1}	F_{B2}	F_{FC2}	T_a	T_{FC3}	T_f	F_{FC3}	F_R
Base Case 4		krypton	0.1	0.2	0.5							0.01
		all others	0.1	$2x10^{-6}$	0.05							$1x10^{-8}$
Base Case 5		krypton	0.1	0.2	0.5	0.9	0.15	300		800		0.1
		cesium	0.1	$2x10^{-6}$	0.05	0.9	$1.6x10^{-3}$	300		800		$9x10^{-4}$
		ruthenium	0.1	$2x10^{-6}$	0.05	0.9	$1.6x10^{-6}$	300		800		$1x10^{-6}$
		particulates	0.1	$2x10^{-6}$	0.05	0.9	0.0	300		800		$5x10^{-8}$
Sensitivity Study 4	Al-U	all	1.0	$2x10^{-7}$	0.05							$1x10^{-8}$
	TRIGA	krypton	1.0	0.2	0.5							0.1
	TRIGA	all others	1.0	$2x10^{-6}$	0.05							$1x10^{-7}$
Sensitivity Study 5A	TRIGA	krypton	1.0	0.2	0.05	1.0	0.15	300		800		0.26
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300		800		0.001
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300		800		$2.3x10^{-6}$
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300		800		$1.3x10^{-6}$
	Al-U	all	1.0	$2x10^{-7}$	0.05	1.0	0.0	300		800		$1.3x10^{-7}$
Sensitivity Study 5B	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300	923	1023	0.33	0.31
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300	923	1023	0.1	0.011
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300	923	1023	0.1	0.0098
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300	923	1023	0.0033	$3.3x10^{-4}$
	Al-U	krypton	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.0	0.098
	Al-U	cesium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	0.1	0.0098
	Al-U	ruthenium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-5}$	$1.7x10^{-6}$
	Al-U	particulates	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.6x10-6	$3.0x10^{-7}$
Sensitivity Study 6A	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300		800		0.35
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300		800		0.0016
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300		800		$3.6x10^{-6}$
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300		800		$2.0x10^{-6}$
	Al-U	all	1.0	$2x10^{-7}$	0.05	1.0	0.0	300		800		$2.0x10^{-7}$
Sensitivity Study 6B	TRIGA	krypton	1.0	0.2	0.5	1.0	0.15	300	923	1023	1.0	1.0
	TRIGA	cesium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-3}$	300	923	1023	0.3	0.3
	TRIGA	ruthenium	1.0	$2x10^{-6}$	0.05	1.0	$1.6x10^{-6}$	300	923	1023	0.3	0.3
	TRIGA	particulates	1.0	$2x10^{-6}$	0.05	1.0	0.0	300	923	1023	0.01	0.01
	Al-U	krypton	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	1.0	1.0
	Al-U	cesium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	0.1	0.1
	Al-U	ruthenium	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-5}$	$1.6x10^{-5}$
	Al-U	particulates	1.0	$2x10^{-7}$	0.05	1.0	0.0	300	923	1023	$1.6x10^{-6}$	$1.8x10^{-6}$

Inspection of Table D5-2 allows the size of the new release fractions developed for aluminum-uranium alloy and TRIGa fuels to be conpared to the release fractions used in the base case calculations. Table D5-5 summarizes these comparisons.

Table D5-5 Relative Size of the Sensitivity Study Release Fractions Compared to the Base Case Release Fractions Used to Perform the Base Case Calculations

Fuel	Severity Category Sensitivity Study	Severity Category Base Case	Size of New Sensitivity Study Release Fractions Compared to Base Case Release Fractions
Aluminum-Uranium Alloy	4	4	About the same (krypton much smaller)
	5A	5	Smaller (cesium 10,000 times smaller)
	5B	5	Cesium 10 times larger
	6A	6	Cesium 5000 times smaller
	6B	6	Cesium 100 times larger
TRIGA	4	4	10 times larger
	5A	5	About the same
	5B	5	Cesium 10 times larger
	6A	6	About the same
	6B	6	Cesium 300 times larger

D5.4 Probability of High-Temperature Events

Data on the temperatures of real ship fires is nearly non-existent. Only one of the five severe fires identified by searching the Lloyd's of London data (Lloyd's, 1991) attained temperatures where steel beams buckled due to thermal stress. Carbon steels begin to soften at about 475°K (395°F) and have lost 90 percent of their strength at about 925°K (1,205°F). Thus, buckling of ship structures due to thermal stress might be expected to occur at about 700°K (800°F), the midpoint of this temperature range, which suggests that one severe fire in five attains temperatures at about 700°K (800°F) and also that P_{T900K} is less than 0.2. Due to the lack of ship board fire temperature data the an attempt has been made to estimate the likelihood of a fire that exceeds 900°K (1160°F).

A shipboard fire can heat the contents of a transportation cask to temperatures above 900°K (1,160°F) only if three conditions are met: (1) the fire must consume a high quality fuel such as gasoline or jet fuel, (2) enough fuel must be available to cause the fire to burn for an hour or more, and (3) the fire cannot be smothered by lack of air or the operation of fire suppression systems. Most severe ships fires involve the burning of the ship's own fuel (bunker or diesel fuel) or of crude oil, when the collision that leads to the fire involves an oil tanker. Thus, P_{T900K}, the chance that a ship fire can heat the contents of a transportation cask to temperatures above 900°K (1,160°F), can be estimated as follows:

$$PT>900K = P_{\text{good fuel}} \times P_{\text{enough fuel}} \times P_{\text{enough oxygen}}$$

Diesel fuel, bunker fuel, and crude oil all have peak flame temperatures that exceed 900°K or 1,160°F (Mudan, 1988), and most polymeric materials (e.g., plastics, wood) have peak flame temperatures of about 1,200°K or 1,700°F (Babrauskas, 1986a). Since fires in cargo holds should behave like enclosure fires, hold fires that burn wood could attain peak temperatures of about 1200°K (1,700°F), if post flashover conditions are attained (Babrauskas, 1986a). So, the fuels and solid materials that are likely to be involved in shipboard fires in cargo holds should be able to heat cask contents to temperatures significantly above

900°K (1,160°F), provided the fire burns long enough and isn't suppressed by lack of oxygen or the operation of fire suppression systems. Thus, $P_{good\ fuel}$, the chance that a long burning hold fire is supported by the burning of a good fuel, is not likely to be small and is here assumed to be 0.9.

The review of ship fires prepared by the French Bureau Veritas for the International Maritime Organization (IMO, 1992) contains data on ship fire durations. Most ship fires (70 to 80 percent) do not burn for an hour. However, most severe ship fires (95 percent) burn for more than an hour. Therefore, the chance that a severe fire involves enough fuel to burn for an hour or more, $P_{enough\ fuel}$, is assumed to be 0.95.

Figure D5-1 presents an event tree for oxygen availability during fires in cargo holds, and is used to estimate $P_{enough\ oxygen}$. The tree shows that most cargo hold fires will be partially starved for oxygen for two reasons, because hold covers will be closed when the fire starts, or will be deliberately closed after it starts in order to smother the fire; or because CO_2 fire suppression systems are installed in the hold and operate successfully. To quantify the event tree provided in Figure D5-1, it was necessary to derive the probability of these two events. The probability that a cargo hold is closed during a collision can be estimated using the following relationship.

$$P_{closed} = 1 - P_{open}$$

where

$$P_{open} = P_{all\ not\ closed} P_{worked} P_{location} \sum (N_i P_{deck}).$$

The derivation of each of the terms in this relationship is described in the following paragraphs.

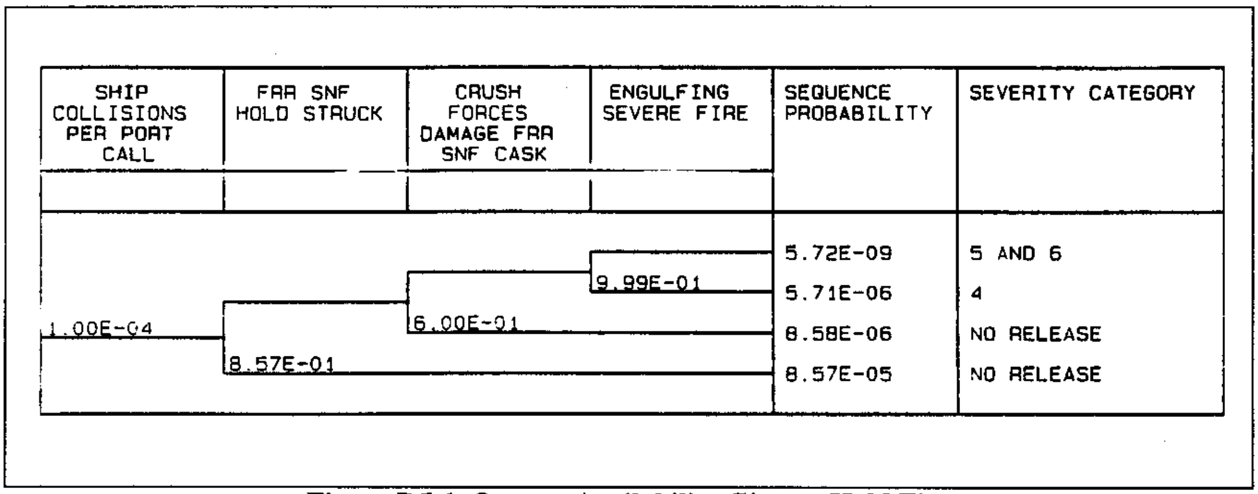


Figure D5-1 Oxygen Availability Given a Hold Fire

Cargo hold covers are normally closed except during loading of unloading of cargo. Thus, if a typical port call takes approximately three days (half a day entering the port and docking, two days anchored at the dock with two-thirds of that time, (two eight-hour shifts per day) spent loading and unloading cargo, and one-half day leaving the port) then all holds will be closed about half of the time while a ship is in port. Conversely, about half of the time at least one hold will be open. Thus $P_{all\ not\ closed} = \frac{1}{2}$ or 0.5.

When a break-bulk freighter like the seven-hold ship used in these analyses is being loaded or unloaded, usually three or four holds are being worked at any given time. Thus, when the ship is being loaded or unloaded, P_{worked}, the probability that a given hold is being worked is ½ or 0.5.

The break-bulk freighter used in these analyses has seven holds. Five of these holds contain three cargo decks, one contains four cargo decks, and one contains only two cargo decks. Thus, there are 21 possible deck locations for a spent fuel cask in this typical ship. Accordingly, $P_{location}$, the chance that a spent nuclear fuel cask has been loaded onto a given deck in one of the seven holds is 0.048.

All hold openings have covers, not just the opening in the main deck through which the hold is loaded and unloaded, but also the openings in the cargo decks within each hold. When a deck in a cargo hold is being loaded or unloaded, all openings above that deck must be open and the opening in the deck and all openings in lower decks are normally closed. Thus, while a hold is being worked, upper decks in that hold will be open to outside air more often than lower decks. For example, for a three-deck hold, while the hold is being worked, the upper deck will always be open to the outside air, the second deck will be open about two-thirds of the time, and the lowest deck will be open about one-third of the time. Thus, if N is the number of holds with two, three, or four decks, and P_{deck} is the probability that deck i in a hold is open to outside air while that hold is being worked, then P_{closed}, the chance that an engulfing fire is partially starved for oxygen because there is a cargo deck or main deck hold cover in place between the fire and the outside air will be:

$$P_{closed} = 1 - \{(0.5)(0.5)(0.048)([5(1 + \tfrac{2}{3} + \tfrac{1}{3})] + [1(1 + \tfrac{3}{4} + \tfrac{1}{2} + \tfrac{1}{4})] + [1(1 + \tfrac{1}{2})])\}$$

$$= 0.833$$

The ORI study (ORI, 1981a) found that over half (60 percent) of all cargo ships are equipped with fire detectors and CO_2 fire suppression systems. Because CO_2 fire suppression systems are not complicated, they should operate reliably on demand most of the time. To be conservative, failed operation during one of five fire events is assumed.

Using this data, the event tree in Figure D5-1 can be quantified to determine the probability of the event $P_{enough\ oxygen}$. Two branches of the oxygen availability tree lead to the outcome "enough air." The probabilities of these two branches sum to 0.087. Thus, 0.09 is a reasonable estimate for $P_{enough\ oxygen}$, the chance that a fire has adequate oxygen available to burn freely and generate maximum heat loads.

Combining the probability estimates for $P_{good\ fuel}$, $P_{enough\ fuel}$, and $P_{enough\ oxygen}$ allows $P_{T900\ K}$ to be estimated as follows:

$$P_{T900\ K} = P_{good\ fuel} \times P_{enough\ fuel} \times P_{enough\ oxygen}$$

$$= 0.9 \times 0.95 \times 0.09 = 0.077$$

Rounding to the next order of magnitude yields a conservative estimate of 0.1 for the chance that a severe engulfing fire with a duration of at least an hour will heat the contents of a transportation cask engulfed by the fire to temperatures significantly higher than 900°K (1,160°F).

D5.5 Probability of Convective Flow through the Failed Cask

Non-uniform heating of the cask during engulfing fires is expected to produce substantial flow of gases through the cask if two or more small holes or one medium hole have been produced in the cask by the ship collision. Because transportation casks bottoms and lid seats are welded to the cylindrical shell of the cask using full-penetration welds that are as strong or stronger than the parent material, when the cask shell is subjected to a severe stress (e.g, high impact or crush forces), the cask shell should yield before the welds fail. In fact, extra-regulatory 60 mph drop tests produced large plastic strains in the cylindrical shell of the test cask without failing its welds (Ludwigsen and Ammerman, 1995). Thus, during a ship collision, crush forces should collapse the cask walls inward without producing catastrophic failure of the lid, its seat, or the welds that attach the seat or the bottom of the cask to the cask walls. Therefore, an unusual configuration of cargo and/or deformed ship structures must be produced during the ship collision in order to subject the cask to forces that will produce failures substantially worse than failure of the lid seal. Either the lid seat must be bent significantly, or at least two penetrations must break, or the cask walls must be sheared or punctured. Although data for such failures is lacking, because casks normally do not fail by these mechanisms, the probability that a failure substantially worse than seal failure occurs is assumed to be no larger than 0.1.

D5.6 Severity Category Event Trees

Figures D5-2 and D5-3 present event trees that represent the sequence of events that lead to category 4, 5A, 5B, 6A, and 6B releases from transportation casks due to ship collisions. After rounding to the nearest integer, Figure D5-3 shows that these categories have the probabilities per port call provided in Table D5-4.

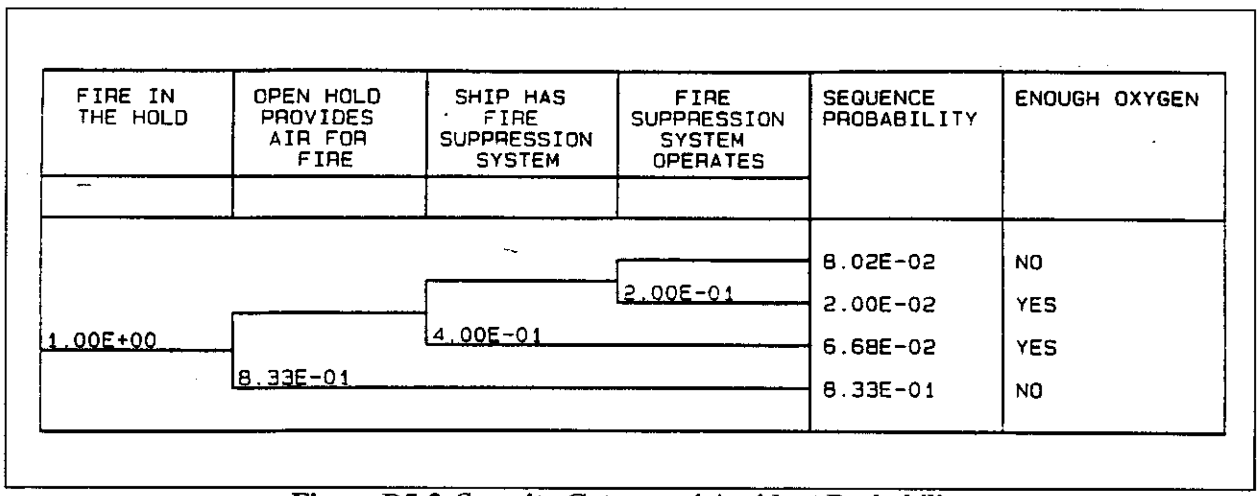


Figure D5-2 Severity Category 4 Accident Probability

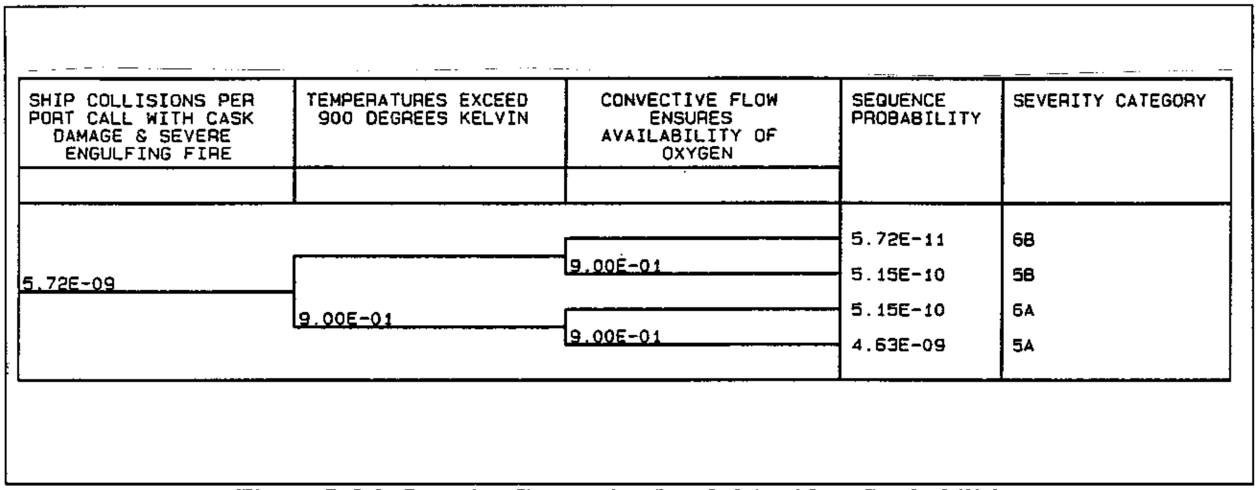


Figure D5-3 Severity Categories 5 and 6 Accident Probabilities

Table D5-4 Sensitivity Study Accident Severity Category Probabilities

Severity Category	*Probability Per Port Call*
4	$6x10^{-6}$
5A	$5x10^{-9}$
5B	$5x10^{-10}$
6A	$5x10^{-10}$
6B	$6x10^{-11}$

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix E
Evaluation of Human Health Effects of Overland Transportation

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

List of Tables

Appendix E
Evaluation of Human Health Effects of Overland Transportation

E.1 Introduction

The overland transportation of any commodity involves a risk to both transportation crew members and members of the public. This risk results directly from transportation-related accidents and indirectly from the increased levels of pollution from vehicle emissions, regardless of the cargo. The transportation of certain materials, such as hazardous or radioactive waste, can pose an additional risk due to the unique nature of the material itself. In order to permit a complete appraisal of the environmental impacts of the proposed action and alternatives, the human health risks associated with the overland transportation of foreign research reactor spent nuclear fuel have been assessed.

This appendix provides an overview of the approach used to assess the human health risks that may result from the overland transportation of foreign research reactor spent nuclear fuel. The appendix includes discussion of the scope of the assessment, analytical methods used for the risk assessment (i.e., computer models), important assessment assumptions, determination of potential transportation routes, and presents the results of the assessment. In addition, to aid in the understanding and interpretation of the results, specific areas of uncertainty are described, with an emphasis on how the uncertainties may affect comparisons of the alternatives.

The approach used in this appendix is modeled after that used in the Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Final Environmental Impact Statement (SNF&INEL Final EIS) (DOE, 1995). The SNF&INEL Final EIS did not perform as detailed an analysis on the specific actions taken for foreign research reactor spent nuclear fuel because of the breadth necessary to analyze the entire spent fuel management program. However, the fundamental assumptions used in this analysis are consistent with those used in the SNF&INEL Final EIS (DOE, 1995), and the same computer codes and generic release and accident data are used.

The risk assessment results are presented in this appendix in terms of "per-shipment" risk factors, as well as for the total risks associated with each alternative. Per-shipment risk factors provide an estimate of the risk from a single spent nuclear fuel shipment between a specific origin and destination. They are calculated for all possible origin and destination pairs for each spent nuclear fuel type. The total risks for a given alternative are found by multiplying the expected number of shipments by the appropriate per-shipment risk factors. This approach provides maximum flexibility for determining the risks for a large number of potential alternatives.

E.2 Scope of Assessment

The scope of the overland transportation human health risk assessment, including the alternatives and options, transportation activities, potential radiological and nonradiological impacts, and transportation modes considered, is described below. Additional details of the assessment are provided in the remaining sections of the appendix.

Proposed Action and Alternatives: The transportation risk assessment conducted for this EIS estimates the human health risks associated with the transportation of spent nuclear fuel for a number of management and implementation alternatives. The alternatives differ primarily in the number and location of possible ports of entry and Phase 1 management sites (storage sites that would be used until a repository was ready). The alternatives considered are described in detail in Chapter 2 of this EIS.

For transportation assessment purposes, each option is defined as an individual or pair of U.S. Department of Energy (DOE) sites used for initial management and an individual or pair of DOE sites used for final interim management. The transportation risk assessment determines risks by considering the total amount of spent nuclear fuel shipped over each representative route. The assessment takes into account differences in the physical and radiological properties of spent nuclear fuel types and characteristics of the potential routes to and between sites.

A large number of potentially applicable marine ports of entry and Canadian border crossings, including commercial and military ports on the Atlantic, Pacific, and Gulf of Mexico coasts are considered in this risk analysis. The port selection process is described in Appendix D. The Canadian border crossing points are representative points based on a qualitative judgment of previously used shipment routes (NRC, 1993). The alternatives in this EIS define the acceptance of the fuel, while the SNF&INEL Final EIS (DOE, 1995) alternative selected defines the DOE site or sites that would receive foreign research reactor spent nuclear fuel; and the options identified in this EIS define the various ways in which the foreign research reactor spent nuclear fuel could be handled to meet the SNF&INEL Final EIS selected alternative.

Transportation-Related Activities: The transportation risk assessment is limited to estimating the human health risks incurred during the overland transportation of spent nuclear fuel for each alternative. The risks to workers or to the public during spent nuclear fuel loading, unloading, and handling prior to or after shipment are not included in the overland transportation assessment, they are addressed in Appendices C and D. Similarly, the transportation risk assessment does not address possible impacts from increased transportation levels on local traffic flow, noise levels, or infrastructure.

Radiological Impacts: For each alternative, radiological risks (i.e., those risks that result from the radioactive nature of the spent nuclear fuel) are assessed for both incident-free (i.e., normal) and accident transportation conditions. The radiological risk associated with incident-free transportation conditions would result from the potential exposure of people to external radiation in the vicinity of a loaded shipment. The radiological risk from transportation accidents would come from the potential release and dispersal of radioactive material into the environment during an accident and the subsequent exposure of people through multiple exposure pathways (i.e., exposure to contaminated ground or air, or ingestion of contaminated food).

All radiological-related impacts are calculated in terms of committed dose and associated health effects in the exposed populations. The radiation dose calculated is the total effective dose equivalent (10 CFR Part 20), which is the sum of the effective dose equivalent (EDE) from external radiation exposure and the 50-yr committed effective dose equivalent from internal radiation exposure. Radiation doses are presented in units of person-rem for collective populations and rem for individuals. The impacts are further expressed as health risks in terms of latent cancer fatalities (LCF) and cancer incidence in exposed populations. The health risk conversion factors (expected health effects per dose absorbed) were derived from International Commission on Radiological Protection Publication 60 (ICRP, 1991).

Nonradiological Impacts: In addition to the radiological risks posed by overland transportation activities, vehicle-related risks are also assessed for nonradiological causes (i.e., related to the transport vehicles and not the radioactive cargo) for the same transportation routes. The nonradiological transportation risks are independent of the radioactive nature of the cargo and would be incurred for similar shipments of any commodity. The nonradiological risks are assessed for both incident-free and accident conditions. Nonradiological risks during incident-free transportation conditions would be caused by potential exposure to increased vehicle exhaust emissions. The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities unrelated to the shipment cargo. State-specific transportation fatality rates are used in the assessment. Nonradiological risks are presented in terms of estimated fatalities.

Transportation Modes: All spent nuclear fuel shipments have been assumed to take place either by truck or rail transportation modes. Per-shipment risk factors are presented separately for truck and rail modes. For the alternatives, risks have been calculated separately for all truck and all rail options, although the actual transportation operation for a selected alternative may involve a combination of the two modes.

Barge transport has certain disadvantages. First, barge transport limits site and port selection for both the SNF&INEL Final EIS and this EIS to Savannah River Site (available both phases) and Hanford Site (available in Phase 2 only). These sites are only served by the ports of Savannah, GA and Portland, OR, respectively. Additionally, barge transportation would require additional intermodal transfers at the port and at the site. At the port, the cask would be removed from the ocean-going vessel and moved by truck to the barge terminal for loading onto a barge. When the barge arrives at the DOE site, the cask would have to be moved to a truck for transport across the site to the receiving basin. Other reasons for not using barge transportation include DOE's lack of extensive experience in shipping casks via barge, the lack of alternative routes, and low speeds. DOE, however, has performed a scoping analysis of barge transportation to assess its relative impacts.

Receptors: Transportation-related risks are calculated and presented separately for workers and members of the general public. The workers considered are truck and rail crew members involved in the actual overland transportation of spent nuclear fuel. The general public includes all persons who could be exposed to a shipment while it is moving or stopped en route. Potential risks are estimated for the collective populations of exposed people, as well as for the hypothetical maximally exposed individual (MEI). The collective population risk is a measure of the radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing various alternatives.

Cumulative Impacts: The cumulative impacts of the transportation of foreign research reactor spent nuclear fuel are calculated and presented as a relative proportion of those described in the SNF&INEL Final EIS (DOE, 1995). The collective dose to the general population and workers is the measure used to quantify cumulative transportation impacts.

E.3 Spent Nuclear Fuel Packaging and Representative Shipment Configurations

Regulations that govern the transportation of radioactive materials are designed to protect the public from the potential loss or dispersal of radioactive materials as well as from routine radiation doses during transit. The primary regulatory approach to ensure safety is through the specification of standards for the packaging of radioactive materials. Because packaging represents the primary barrier between the radioactive material being transported and radiation exposure to the public and the environment, packaging requirements are an important consideration for the transportation risk assessment. Regulatory packaging requirements are discussed briefly below and in Chapter 5. In addition, the representative packaging and shipment configurations assumed for this EIS are described.

E.3.1 Packaging Overview

Although several Federal and State organizations are involved in the regulation of radioactive waste transportation, primary regulatory responsibility resides with the U.S. Department of Transportation and the U.S. Nuclear Regulatory Commission (NRC). All transportation activities must take place in accordance with the applicable regulations of these agencies specified in 49 Code of Federal Regulations (CFR) Part 173 and 10 CFR Part 71.

Transportation packaging for radioactive materials must be designed, constructed, and maintained to ensure that the packages will contain and shield their contents during normal transport conditions. For more highly radioactive material, such as spent nuclear fuel, they must contain and shield their contents in the event of severe accident conditions. The type of packaging used is determined by the total radioactive hazard presented by the material within the packaging. The basic types of packaging required by the applicable regulations are designated as Type A, Type B, or "strong and tight".

"Strong and tight" packages are designed such that no radioactive material will leak or be released during transportation. They can only be used for low-specific-activity material. Type A packaging must withstand the conditions of incident-free transportation without the loss or dispersal of the radioactive contents. Incident-free transportation refers to all conditions of transportation except those that result from accidents or sabotage. Approval of Type A packaging is achieved by demonstrating that the packaging can withstand specified test conditions which are intended to simulate incident-free transportation conditions. Type A packaging, typically a 55-gallon (gal) drum or standard waste box, is commonly used to transport wastes having low activities of radioactive material.

The transportation of spent nuclear fuel requires the use of Type B packaging. In addition to meeting the standards for Type A packaging, Type B packaging must provide a high degree of assurance that even in severe accidents the integrity of the package will be maintained with essentially no loss of the radioactive contents or serious impairment of the shielding capability. Type B packaging must satisfy stringent testing criteria specified in 10 CFR Part 71. The testing criteria were developed to simulate severe hypothetical accident conditions, including impact, puncture, fire, and water immersion. The massive casks used to transport spent nuclear fuel represent the most widely recognized Type B packaging.

For risk assessment purposes, it is important to note that all packaging of a given type is designed to meet the same performance criteria. Therefore, two spent nuclear fuel casks of different designs would be expected to perform similarly during incident-free and accident transportation conditions. The specific cask selected, however, will determine the total number of shipments necessary to transport a given quantity of spent nuclear fuel.

External radiation allowed to escape from a package must be below specified limits that minimize the exposure of the handling personnel and general public. The foreign research reactor spent nuclear fuel shipments would be handled only by the shipper and the receiver, an arrangement referred to as an "exclusive-use" shipment. For these types of shipments, the external radiation dose rate during normal transportation conditions must be maintained below the following limits of 49 CFR Part 173:

- 10 mrem per hr at any point 2 meters (m) (6.6 ft) from the vertical planes projected by the outer lateral surfaces of the transport vehicle (referred to as the regulatory limit throughout this document), and
- 2 mrem per hr in any normally occupied position in the transport vehicle.

Although additional restrictions apply to package surface radiation levels, these restrictions are not important for the transportation radiological risk assessment.

The NRC recently issued revised regulations, 10 CFR Part 71, governing the transportation of radioactive materials. These regulations become effective on April 1, 1996 (NRC, 1995). The revised regulations conform with those of the International Atomic Energy Agency and current legislative requirements. The revised regulations affecting "Type B" casks require that a spent nuclear fuel transportation cask with activity greater than one million curies (Ci) be designed and 290 psi, or immersion in 200 m (656 ft) of water, for a period of not less than one hour without collapse, buckling, or allowing water to leak into the cask.

E.3.2 Packaging and Representative Shipment Configurations for Foreign Research Reactor Spent Nuclear Fuel

To conduct the overland transportation risk assessment, assumptions must be made concerning the types of packaging, transport vehicles, and shipment capacities that could be used for future spent nuclear fuel shipments. In all cases, it is assumed that spent nuclear fuel would be characterized, treated, packaged, and labeled in accordance with applicable regulations prior to shipment.

The transportation of all foreign research reactor spent nuclear fuel would take place in casks certified by foreign competent authorities and revalidated by Department of Transportation in accordance with 49 CFR 173. In addition, it is assumed that only exclusive-use vehicles would be used. Highway transportation is assumed to take place by legal weight heavy-haul combination (tractor-trailer) trucks. Rail transportation is assumed to take place by regular freight train service.

E.3.3 Description of Transportation Activities

The proposed action could involve transporting foreign research reactor spent nuclear fuel from the ports of entry (both marine ports and Canadian border crossings) to DOE sites, and could involve transporting foreign research reactor spent nuclear fuel between DOE sites. The interim management site or sites for the foreign research reactor spent nuclear fuel in the United States have been determined on the basis of the SNF&INEL Final EIS (DOE, 1995).

In this section, the assumptions and logic used to model the transportation requirements for the basic implementation of Management Alternative 1 of the proposed action are described. In general, the same assumptions are used to analyze the management and implementation alternatives. Therefore, the transportation requirements for management and implementation alternatives will be described in relation to the basic implementation.

Certain assumptions are required in order to simply and consistently describe the manner in which foreign research reactor spent nuclear fuel would be transported to the sites. The shipments were divided into east coast and west coast shipments, depending on the country of origin. Spent nuclear fuel shipments from Europe, Africa, the Middle East and parts of South and Central America were designated as east coast shipments, and all others were designated as west coast shipments. Shipments from Canada were assumed to enter the United States from either an eastern or western point of entry, depending on the Canadian point of origin. Under these assumptions, the east coast would receive approximately 535 cask shipments and the west coast approximately 186 cask shipments. Approximately 116 shipments from Canada would arrive in the eastern United States.

Regarding foreign research reactor spent nuclear fuel transportation, the SNF&INEL Final EIS (DOE, 1995) analyzes the use of any one of five candidate sites and seven distinct combinations of sites. Eight of the alternatives involve sites that could not be ready to accept spent nuclear fuel at the onset of the foreign research reactor spent nuclear fuel program. Therefore, a two-phased approach is assumed using one or both of the sites that are ready to accept spent nuclear fuel (Savannah River Site and Idaho National Engineering Laboratory) as a near-term management location. Phase 1 is defined, for the purposes of analyzing transportation, as the period of time in which shipments of foreign research reactor spent nuclear fuel are transported to a near-term management site. For analytical purposes, Phase 1 is assumed to last from the beginning of 1996 to the beginning of 2006.

The amount of fuel that would arrive in Phase 1 versus Phase 2 cannot be precisely determined at this time. In order to proceed with the risk analysis, it is necessary to make assumptions based on the available information. The total number of casks that would be required to transport the 22,700 spent fuel elements is estimated to be 837, per Appendix B. The split between Phase 1 and Phase 2 depends on the rate at which casks are received and the time the Phase 2 site(s) is ready to receive fuel. For calculational purposes, the casks are assumed to arrive at a uniform rate, and the Phase 2 site(s) is assumed to be ready 10 years after the implementation of the policy.

The disposition of foreign research reactor spent nuclear fuel during Phase 1 is analyzed in this EIS. Logically, Phase 1 could entail any one of four options: A) splitting foreign research reactor spent nuclear fuel by fuel type [TRIGA (which stands for Training, Research, and Isotope reactors built by General Atomic) to Idaho National Engineering Laboratory and Aluminum-based to Savannah River Site], B) splitting the spent nuclear fuel geographically by port of entry, C) transporting all spent nuclear fuel to Idaho National Engineering Laboratory, or D) transporting all spent nuclear fuel to Savannah River Site. Not all Phase 1 strategies are consistent with all Phase 2 strategies.

Phase 2 begins when Oak Ridge Reservation, Hanford Site, or Nevada Test Site would be ready to receive fuel from ports and, when applicable, from a DOE site being used for near-term management. In all cases, Phase 2 is dependent on decisions based on the SNF&INEL Final EIS (DOE, 1995). During Phase 2, all foreign research reactor spent nuclear fuel arriving at ports of entry would be transported to the appropriate site. Additionally, intersite shipments from the near-term management site could also be arriving at the SNF&INEL Final EIS selected site(s).

The following is a description of the shipping program, organized by SNF&INEL Final EIS (DOE, 1995) alternatives:

No Action - DOE cannot accept foreign research reactor spent nuclear fuel under this alternative.

Decentralization - Foreign research reactor spent nuclear fuel arriving on the east coast would be transported to Savannah River Site, and foreign research reactor spent nuclear fuel arriving on the west coast would be transported to Idaho National Engineering Laboratory. Since both Idaho National Engineering Laboratory and Savannah River Site are capable of receiving fuel in late 1995, there is no need for a two-phase program or intersite shipments. The total number of shipments for this alternative would be approximately 837. Savannah River Site would receive 651 casks from the east, and Idaho National Engineering Laboratory would receive 186 casks from the west. The transportation under this alternative is illustrated in Figure E-1. No intersite shipment would be anticipated under this single-phased alternative.

Figure E-1 Decentralization: Spent Nuclear Fuel to Idaho National Engineering Laboratory and Savannah River Site

1992-1993 Planning Basis - The SNF&INEL Final EIS (DOE, 1995) provides no specific guidance for foreign research reactor spent nuclear fuel. The transportation analysis in the SNF&INEL Final EIS assumed that half the foreign research reactor spent nuclear fuel would be transported to Idaho National Engineering Laboratory and half to Savannah River Site. The disposition of foreign research reactor spent nuclear fuel could correspond to Decentralization (described above), Regionalization (described below), Centralization to Idaho National Engineering Laboratory or Savannah River Site (described below), or an arbitrary split as described in the SNF&INEL Final EIS (DOE, 1995).

Regionalization - There are two distinct subalternatives under Regionalization: Regionalization by Fuel Type, and Regionalization by Geography. These subalternatives are described below.

Regionalization Subalternative A - Under Regionalization by Fuel Type, the foreign research reactor spent nuclear fuel would be split by fuel type, regardless of the port of entry. The TRIGA fuel would be shipped to Idaho National Engineering Laboratory and the aluminum-based Material Test Reactor (MTR) fuel would be shipped to Savannah River Site. Savannah River Site would receive 675 casks of fuel: 544 from the east and 131 from the west. Idaho National Engineering Laboratory would receive 162 casks of fuel: 107 from the east and 55 from the west. The transportation under this alternative is illustrated in Figure E-2. No intersite shipment would be anticipated under this single-phased alternative.

Regionalization Subalternative B - Under Regionalization by Geography, foreign research reactor spent nuclear fuel would be distributed between an Eastern Regional Site (Oak Ridge Reservation or Savannah River Site) and a Western Regional Site (either Hanford Site, Idaho National Engineering Laboratory, or Nevada Test Site). The foreign research reactor spent nuclear fuel arriving at an eastern port would go to the Eastern Regional Site, and the foreign research reactor spent nuclear fuel arriving at a western port would go to the Western Regional Site. If the chosen sites were Savannah River Site and Idaho National

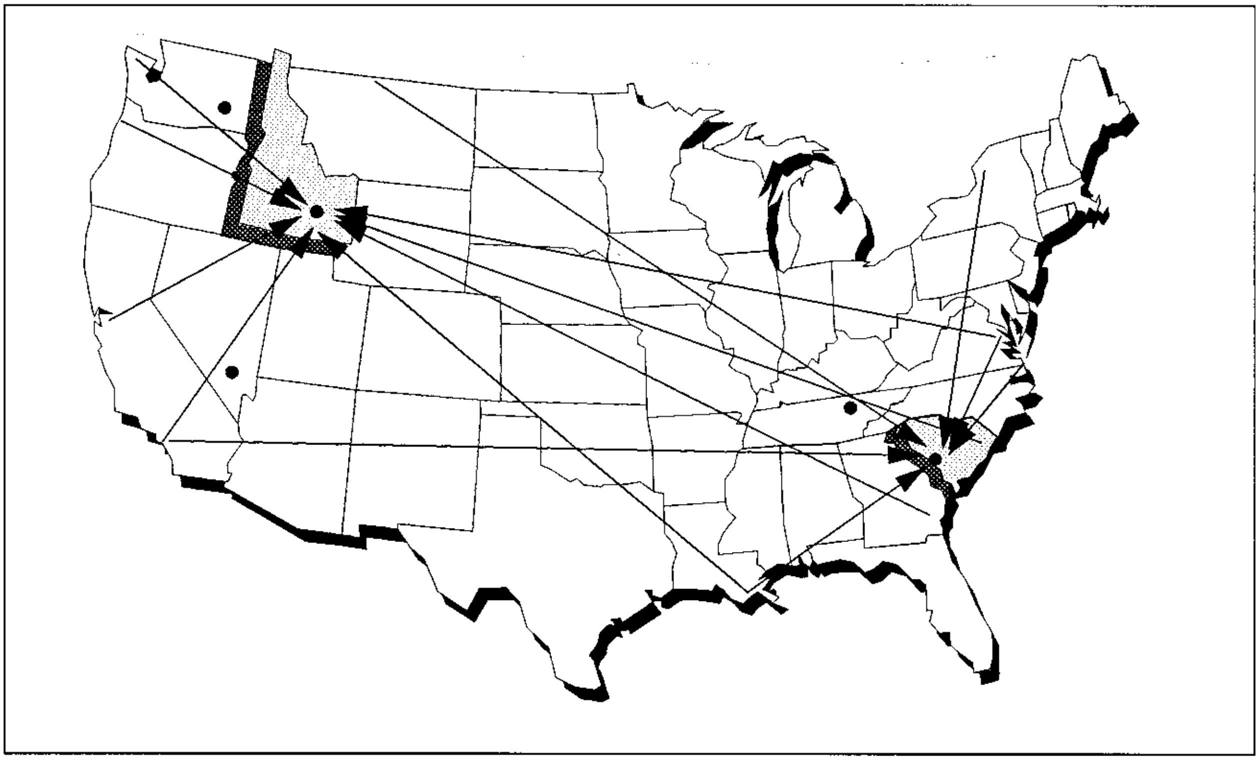


Figure E-2 1992/1993 Regionalization by Fuel Type: TRIGA Spent Nuclear Fuel to Idaho National Engineering Laboratory and MTR Spent Nuclear Fuel to Savannah River Site

Engineering Laboratory, the transportation would be the same as that described in the Decentralization Alternative and Figure E-2. No intersite shipment would be anticipated under this single-phased alternative.

A two-phased program would be required if a site other than Idaho National Engineering Laboratory or Savannah River Site were selected as a regional site under this programmatic alternative. The remaining possible site pairs for Regionalization are Idaho National Engineering Laboratory/Oak Ridge Reservation, Nevada Test Site/Savannah River Site, Nevada Test Site/Oak Ridge Reservation, Hanford Site/Savannah River Site, and Hanford Site/Oak Ridge Reservation. Splitting fuel by both geography and fuel type was considered as a logical Phase 1 approach for each site pair, but transporting all fuel to Savannah River Site or Idaho National Engineering Laboratory for near-term management was not considered for the following reasons:

- If Idaho National Engineering Laboratory were selected as the Western Regional Site, and Savannah River Site were not selected as the Eastern Regional Site, it would not be reasonable to ship all foreign research reactor spent nuclear fuel to Savannah River Site during Phase 1. Since Idaho National Engineering Laboratory is currently capable of receiving fuel and is much closer to west coast ports, it would be unreasonable to ship all fuel across the country to Savannah River Site only to move the fuel again. However, the option to ship all MTR fuel to Savannah River Site, for onsite logistical reasons, is a logical Phase 1 option even if Savannah River Site is not an ultimate interim management location. Thus, shipment of all fuel to SRS during Phase 1 was not considered reasonable if Idaho National Engineering Laboratory were to be chosen as the Western Regional Site.

- Conversely, if Savannah River Site were selected as the Eastern Regional Site, and Idaho National Engineering Laboratory were not selected as the Western Regional Site, it would not be reasonable to ship all foreign research reactor spent nuclear fuel to Idaho National Engineering Laboratory during Phase 1. Since Savannah River Site is currently capable of receiving fuel and is much closer to east coast ports, it would be unreasonable to ship all fuel across the country to Idaho National Engineering Laboratory only to move the fuel again. However, the option to ship all TRIGA fuel to Idaho National Engineering Laboratory, for onsite logistical reasons, is a logical Phase 1 option, even if Idaho National Engineering Laboratory is not an ultimate interim management location. Thus, shipment of all fuel to Idaho Engineering National Laboratory during Phase 1 was not considered reasonable if Savannah River Site were to be chosen as the Eastern Regional Site.

Figures E-3 through E-7 show the transportation schemes for site pairs Idaho National Engineering Laboratory/Oak Ridge Reservation, Nevada Test Site/Savannah River Site, Nevada Test Site/Oak Ridge Reservation, Hanford Site/Savannah River Site, and Hanford Site/Oak Ridge Reservation, respectively. The origins of the arrows representing shipments on the figures are selected for illustrative purposes, not to show specifically selected ports. Shipments would be expected to arrive at eastern, western, and Gulf Coast ports, and from eastern and western Canada. Because of their relative proximity to eastern sites, Gulf Coast ports are assigned the same transportation schemes as east coast ports. Note that there is no TRIGA fuel in Canada, so there is no planned route from Canada to Idaho National Engineering Laboratory for the Regionalization by Fuel Type alternative.

The number of shipments in the basic implementation for each site pair is described in Table E-1. The number of intersite shipments is based on the assumption that spent nuclear fuel arriving from foreign countries in small casks would be rearranged such that intersite shipments could be made in larger casks. The fuel assemblies would be cut to more efficient shapes, the fuel would be older and, thus, less radioactive and would produce less heat. For analysis purposes, it is assumed that the amount of foreign research reactor spent nuclear fuel originally shipped in four of the casks used for importing fuel could be shipped in one intersite truck cask. Rail shipment allows the use of even larger casks; and, thus, it is assumed that 10 cask loads of foreign research reactor spent nuclear fuel could be intersite shipped in 1 rail cask. Since the potential shipments would be scheduled to occur at least 10 years in the future, it is difficult to predict what casks would be used. Appendix B describes a variety of candidate casks. DOE would use fewer but larger shipments when shipping from site-to-site.

Centralization - Any one of the five DOE sites could be chosen by the SNF&INEL Final EIS (DOE, 1995) for receipt of all foreign research reactor spent nuclear fuel. If that site were Idaho National Engineering Laboratory or Savannah River Site, a two-phased approach would not be necessary. From the beginning of the program, all fuel could be accepted at either of these sites. Figures E-8 and E-9 describe the single-phase centralization options to Idaho National Engineering Laboratory and Savannah River Site, respectively.

However, a two-phase program would be required if the site selected were Hanford Site, Oak Ridge Reservation, or Nevada Test Site, none of which would be ready to receive foreign research reactor spent nuclear fuel at the beginning of the program. The option for execution of Phase 1 shipments is assumed to be independent from Phase 2 and the chosen SNF&INEL Final EIS (DOE, 1995) alternative. As with the Regionalization options, during Phase 1, DOE could choose to divide the fuel by either geography or fuel type between the two initially-capable DOE sites (Idaho National Engineering Laboratory and Savannah River Site). Alternatively, all fuel could be initially shipped to Idaho National Engineering Laboratory or Savannah River Site. Figures E-8 through E-12 show the transportation schemes for all five sites. The

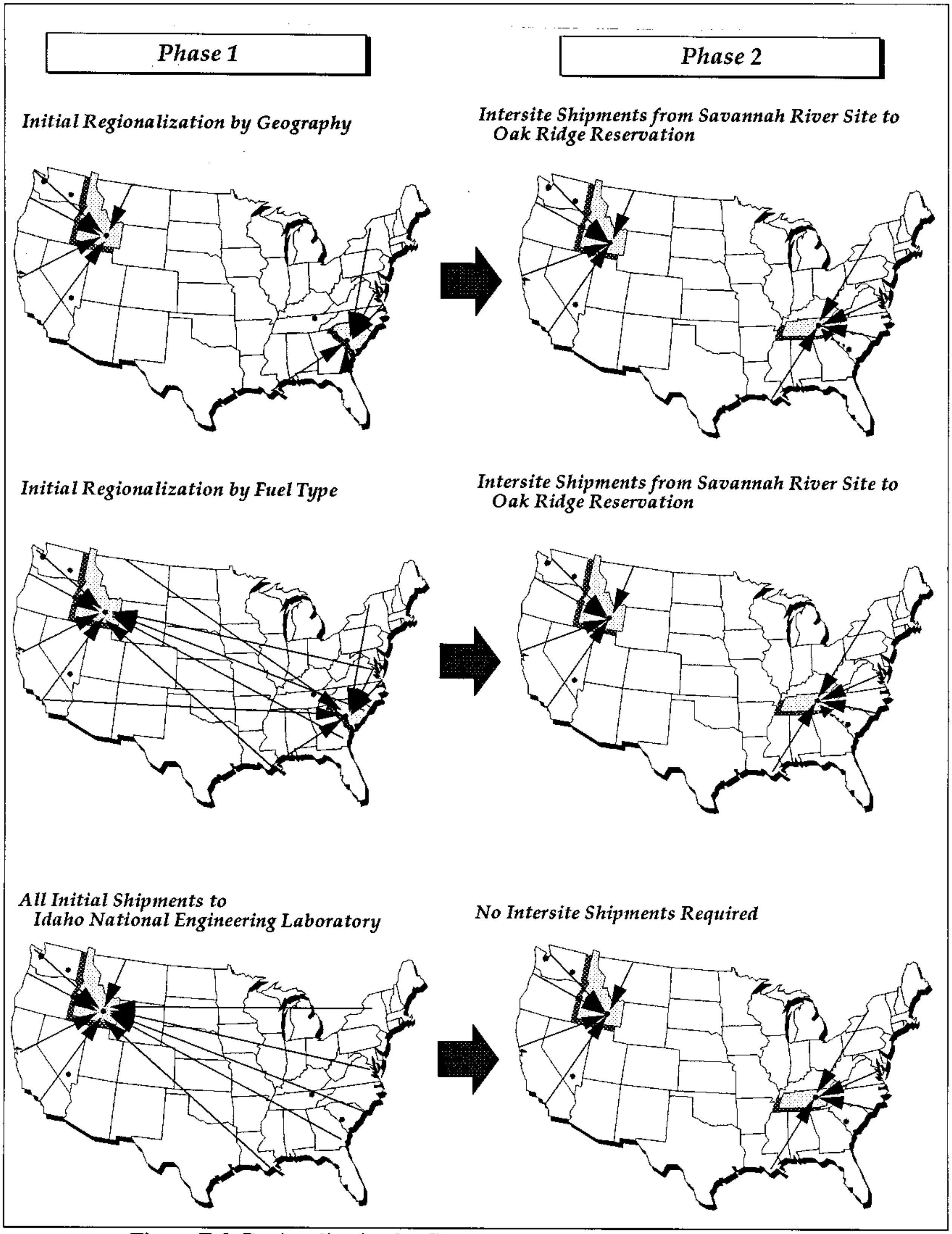


Figure E-3 Regionalization by Geography to Idaho National Engineering Laboratory and Oak Ridge Reservation

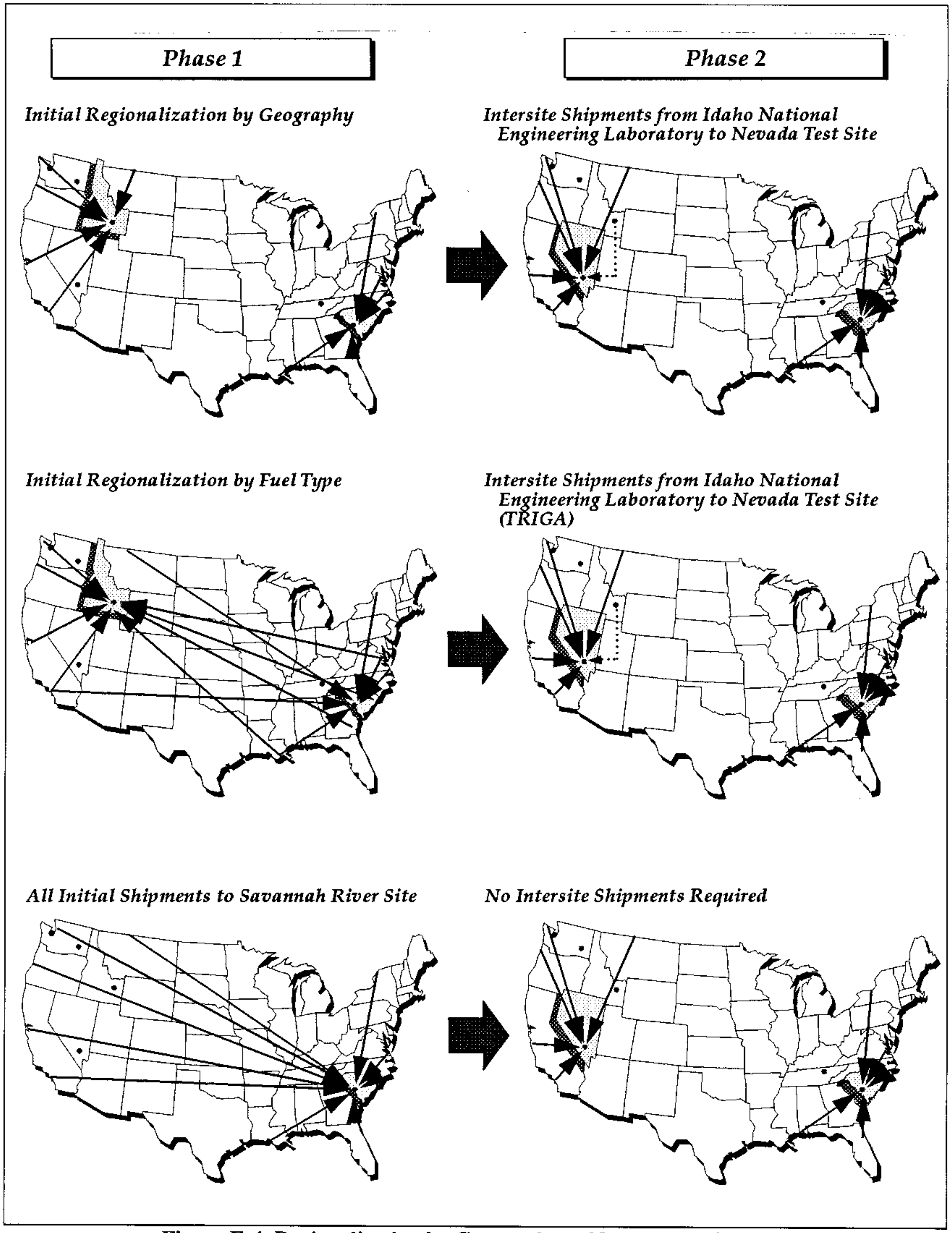


Figure E-4 Regionalization by Geography to Nevada Test Site and Savannah River Site

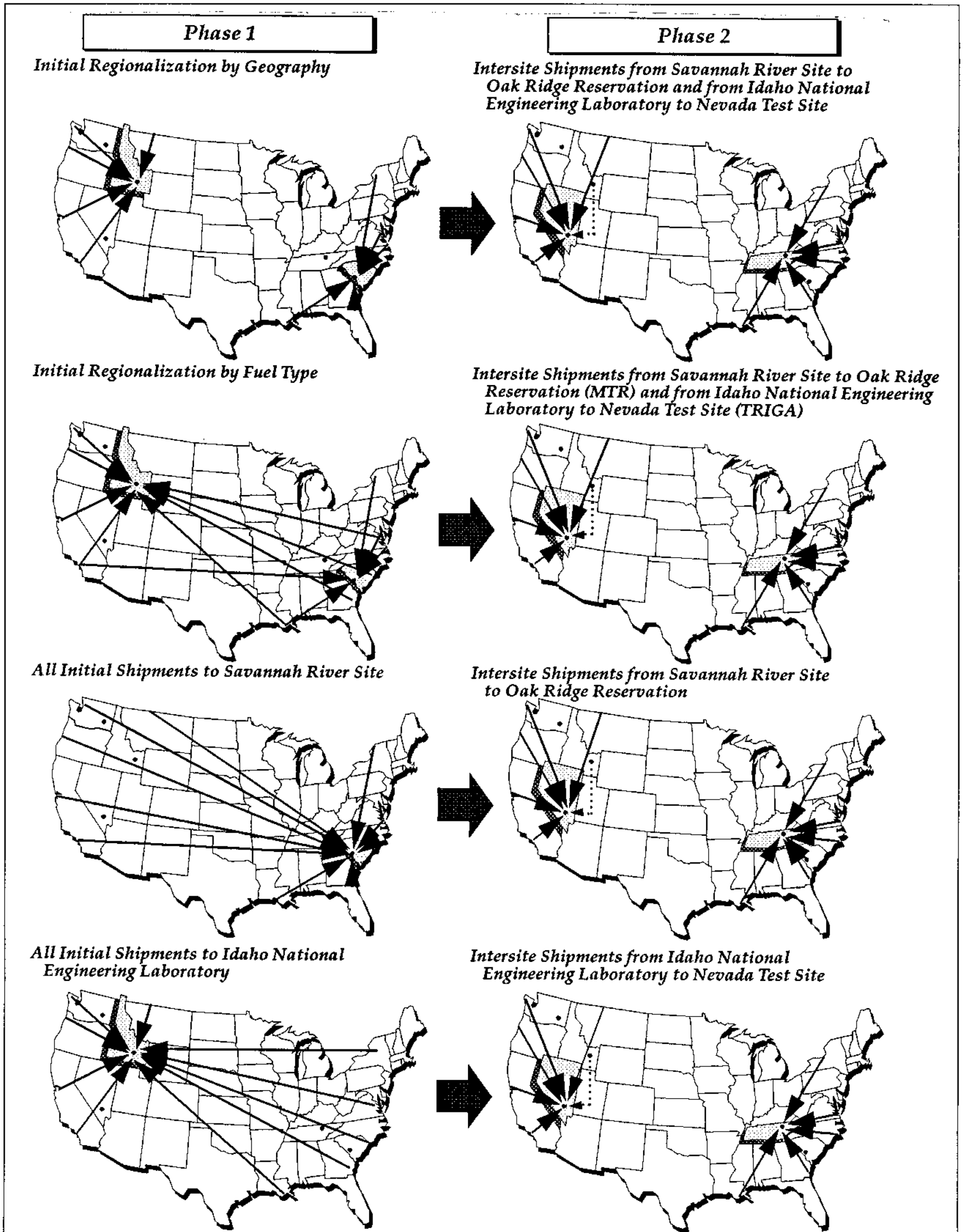


Figure E-5 Regionalization by Geography to Nevada Test Site and Oak Ridge Reservation

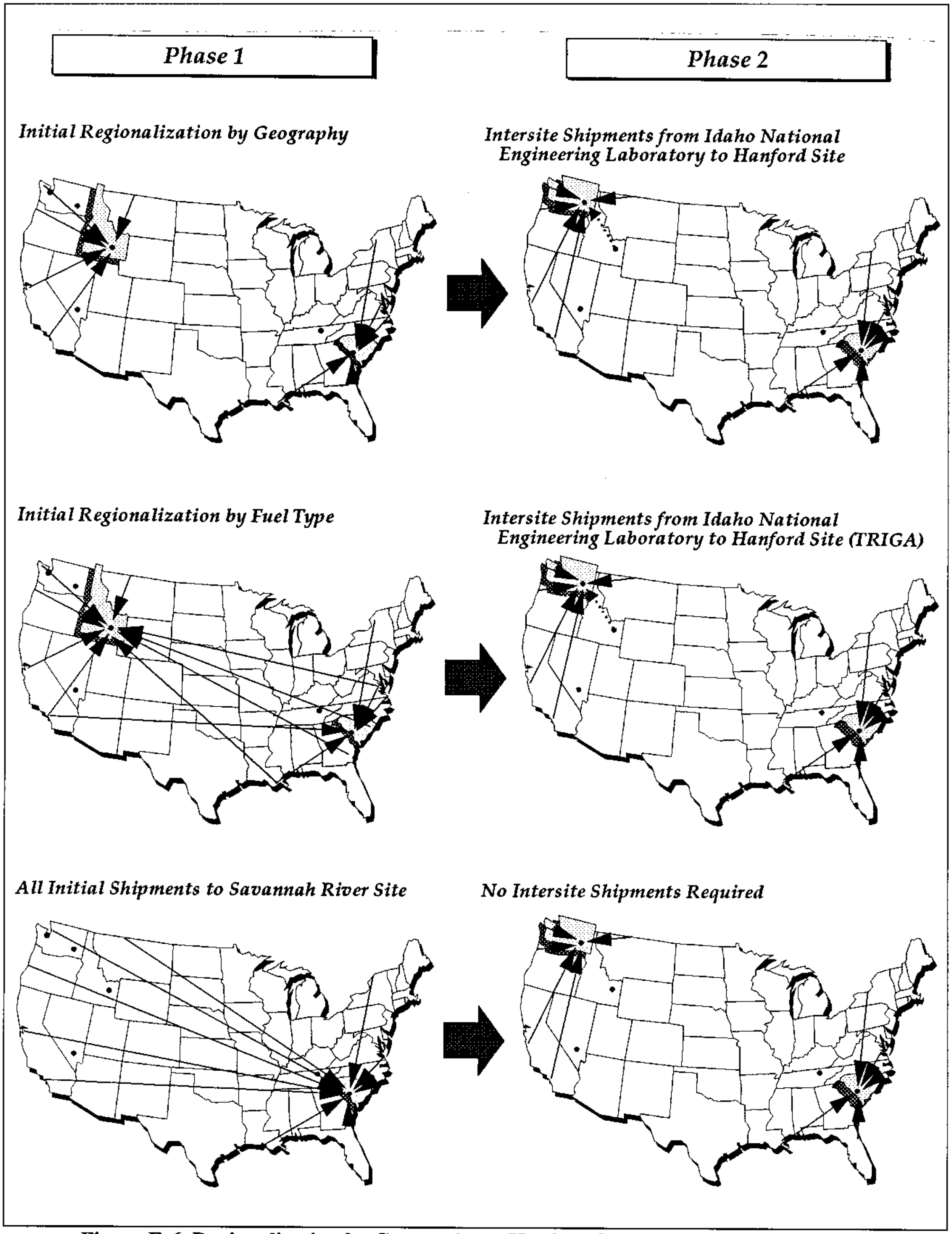


Figure E-6 Regionalization by Geography to Hanford Site and Savannah River Site

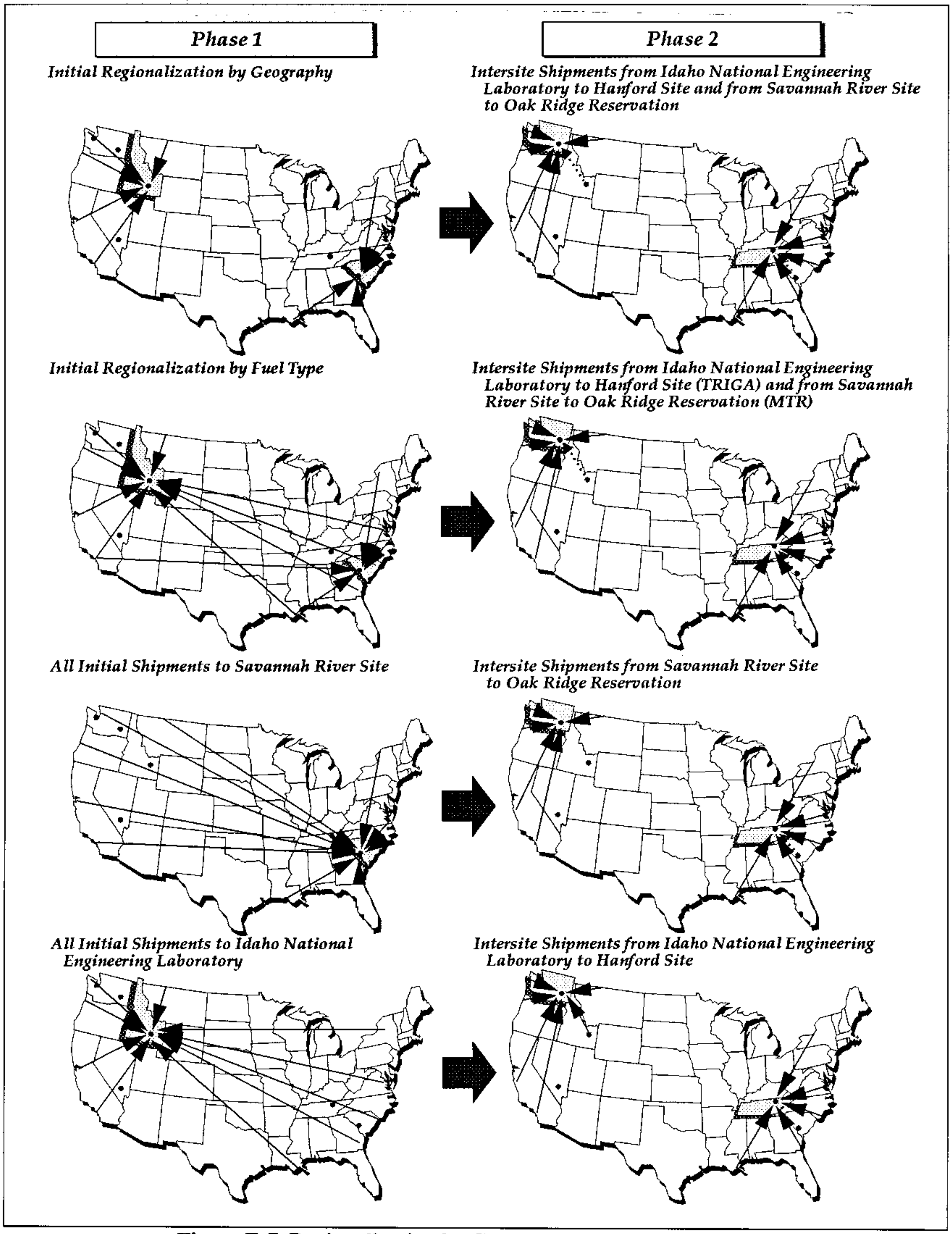


Figure E-7 Regionalization by Geography to Hanford Site and Oak Ridge Reservation

Table E-1 Shipment Summary for Regionalization Alternatives

Spent Nuclear Fuel Site Option	Phase 1 Approach	Phase 1 Port-to-Site Shipments	Site-to-Site Shipments[a]	Phase 2 or Port-to-Final Site Shipments	Total Number of Shipments
INEL/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51	East to ORR: 150 West to INEL: 43	963/888
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52	East to ORR: 150 West to INEL: 43	967/889
	All to INEL	644	None	East to ORR: 150 West to INEL: 43	837
NTS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to NTS: 36/15	East to SRS: 150 West to NTS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to NTS: 31/13	East to SRS: 150 West to NTS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to NTS: 43	837
NTS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to NTS: 36/15	East to ORR: 150 West to NTS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to NTS: 31/13	East to ORR: 150 West to NTS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to NTS: 43	998/902
	All to INEL	644	INEL to NTS: 161/65	East to ORR: 150 West to NTS: 43	998/902
HS/SRS	Geographic	East to SRS: 501 West to INEL: 143	INEL to HS: 36/15	East to SRS: 150 West to HS: 43	873/852
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	INEL to HS: 31/13	East to SRS: 150 West to HS: 43	868/850
	All to SRS	644	None	East to SRS: 150 West to HS: 43	837
HS/ORR	Geographic	East to SRS: 501 West to INEL: 143	SRS to ORR: 126/51 INEL to HS: 36/15	East to ORR: 150 West to HS: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	SRS to ORR: 130/52 INEL to HS: 31/13	East to ORR: 150 West to HS: 43	998/902
	All to SRS	644	SRS to ORR: 161/65	East to ORR: 150 West to HS: 43	998/902
	All to INEL	644	INEL to HS: 161/65	East to ORR: 150 West to HS: 43	998/902

[a] *Truck/Rail shipments, assuming that the truck casks used for intersite shipments are capable due to consolidation of carrying 4 times as much fuel, and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor.*

INEL = Idaho National Engineering Laboratory; ORR = Oak Ridge Reservation; SRS = Savannah River Site; NTS = Nevada Test Site; HS = Hanford Site

number of shipments for each site pair is shown in Table E-2. The number of intersite shipments is based on a 4-cask-to-1 conversion if trucks were used, and a 10-cask-to-1 conversion if trains were used, as explained in the previous section.

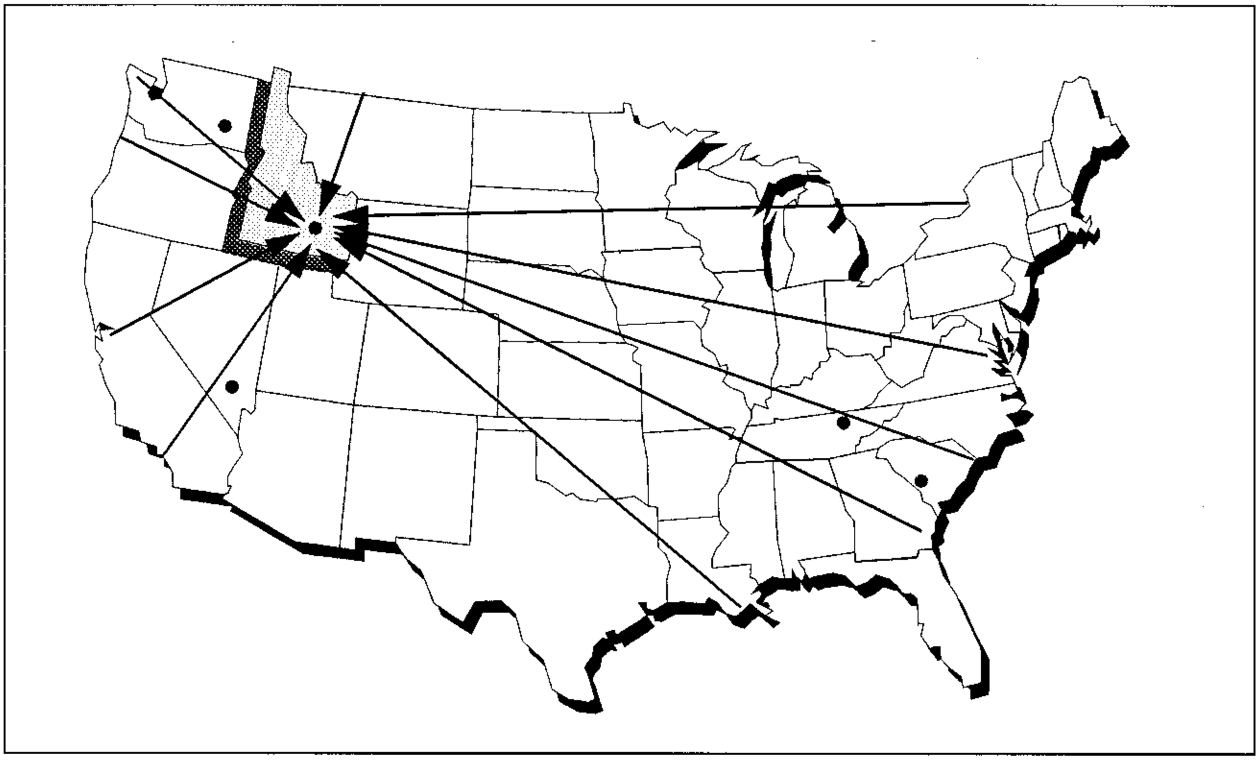


Figure E-8 Centralization to Idaho National Engineering Laboratory

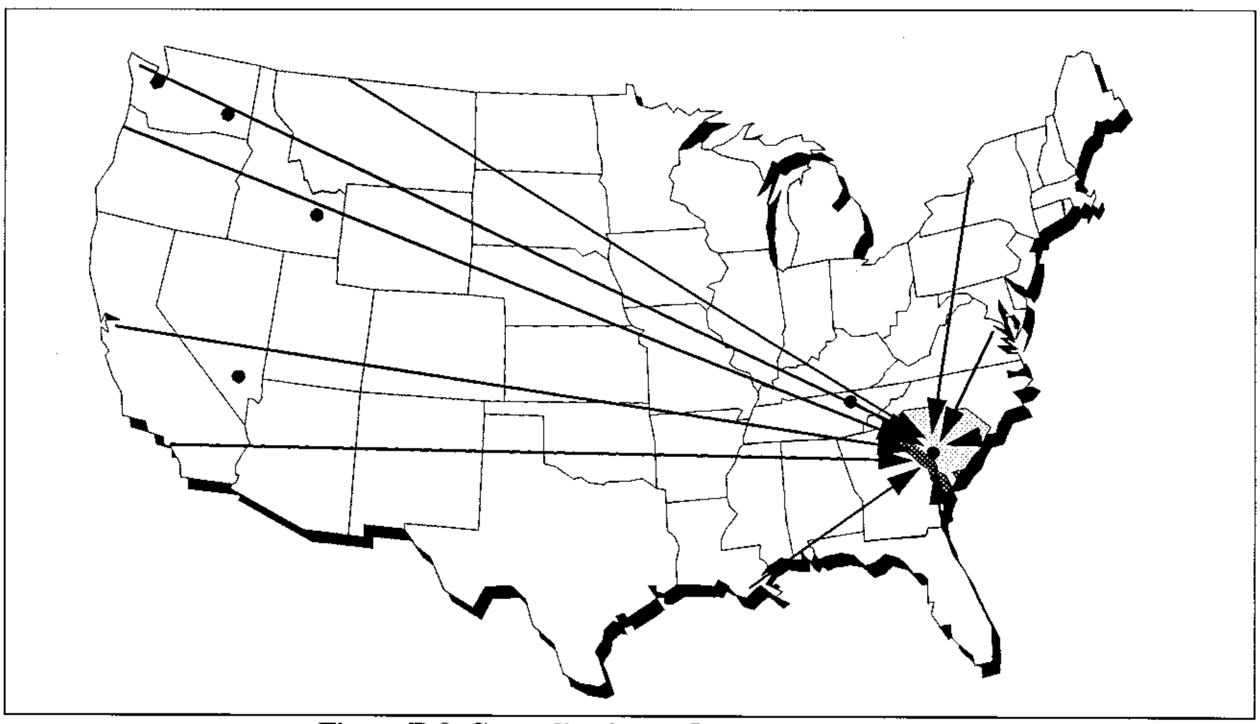


Figure E-9 Centralization to Savannah River Site

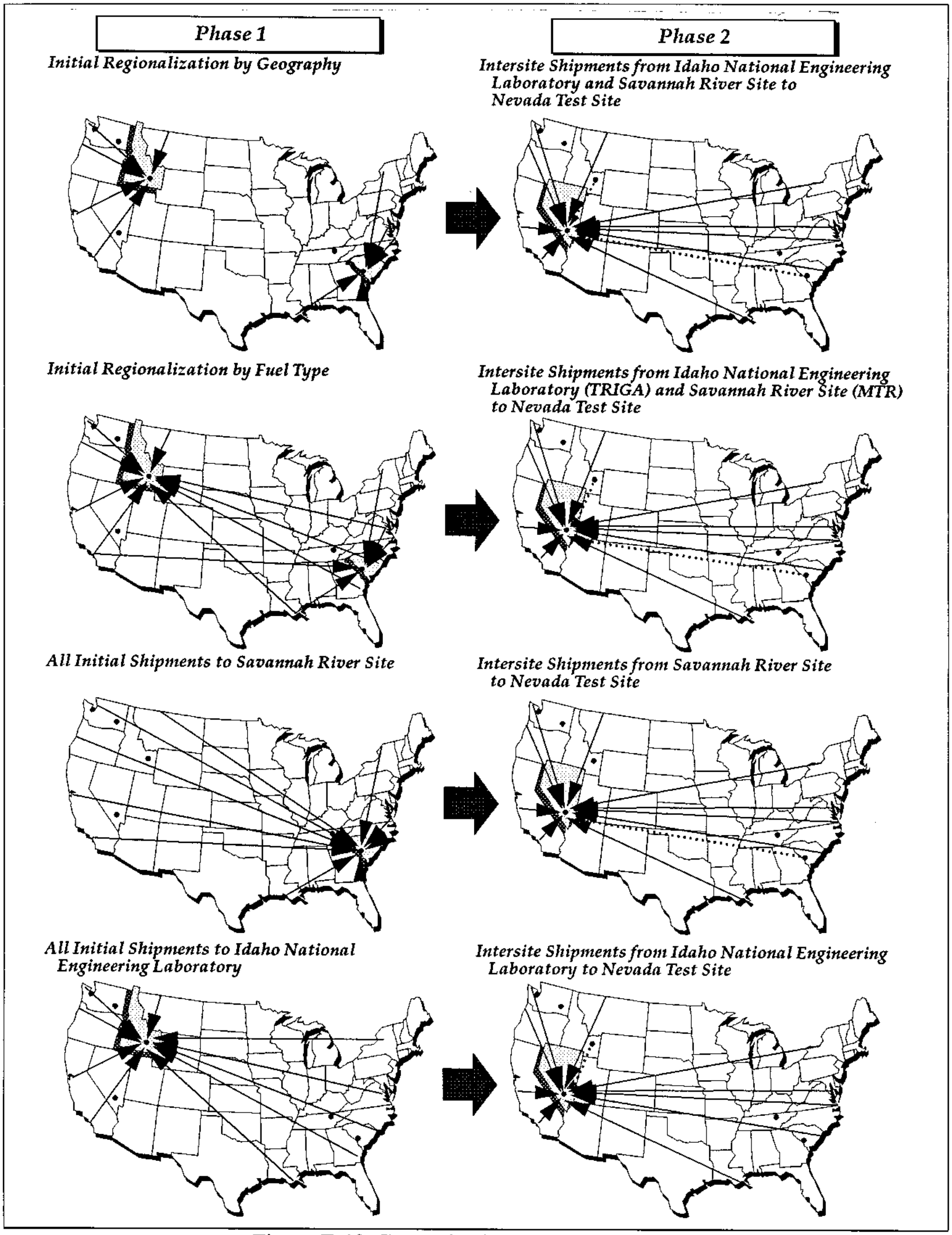


Figure E-10 Centralization to Nevada Test Site

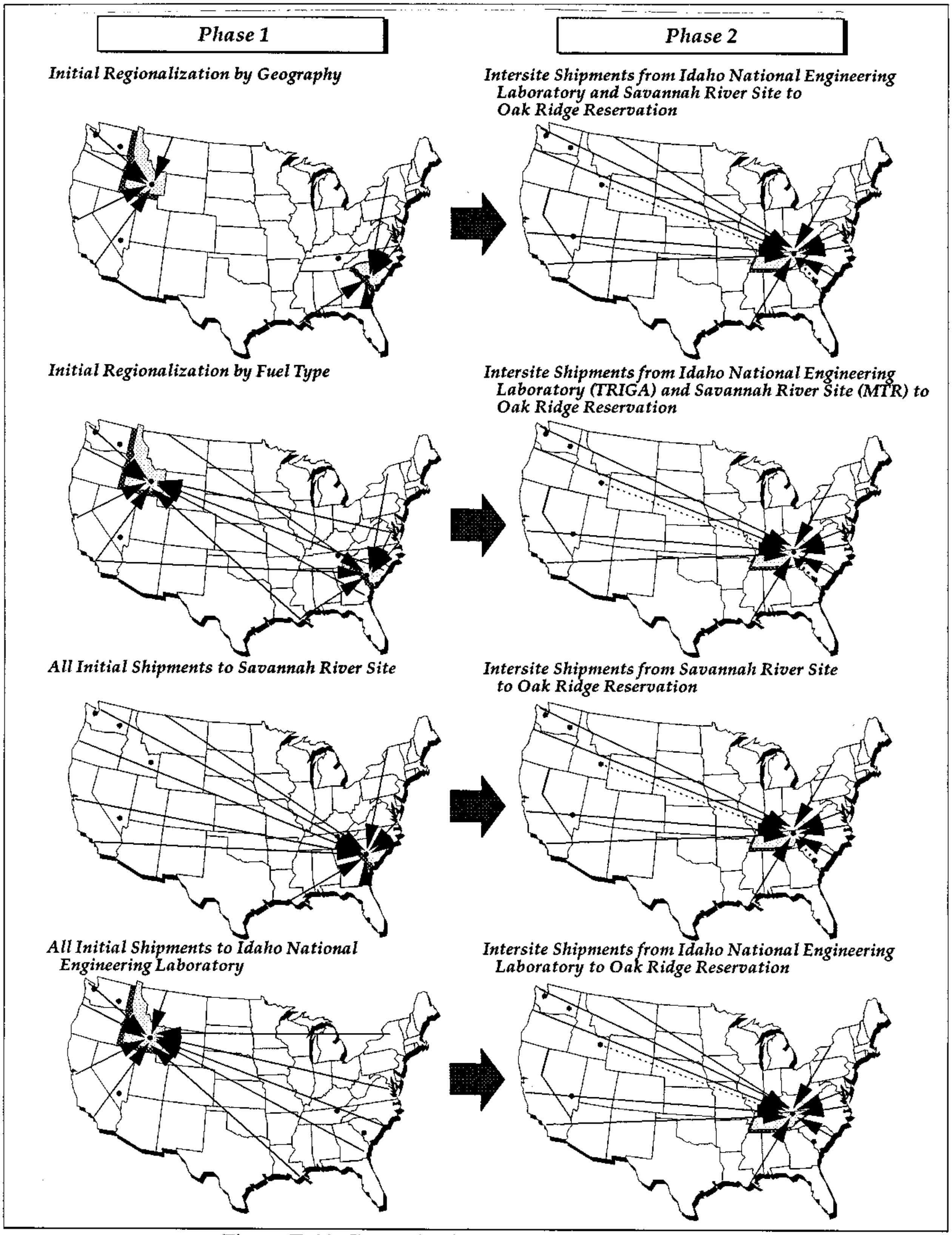


Figure E-11 Centralization to Oak Ridge Reservation

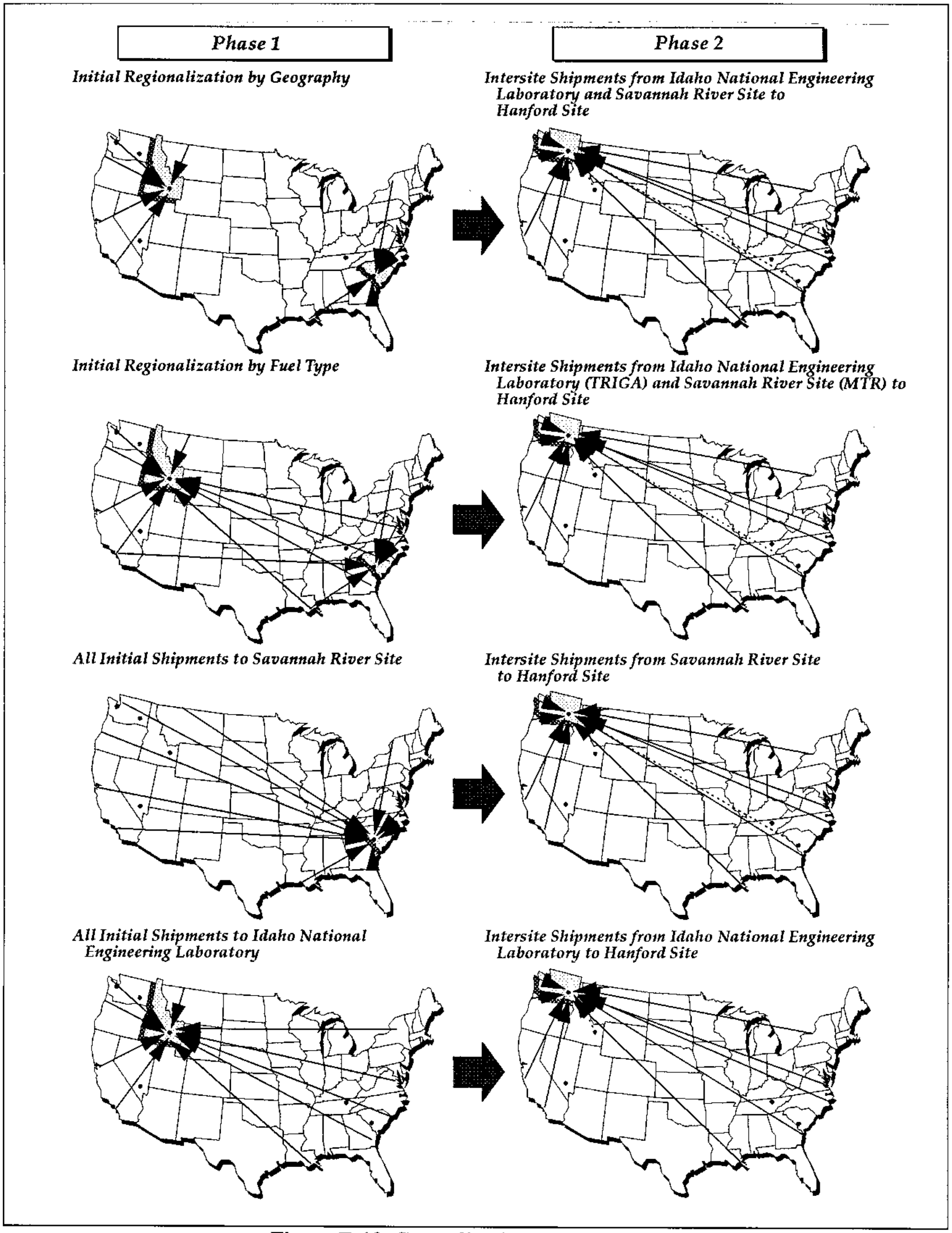


Figure E-12 Centralization to Hanford Site

Table E-2 Shipment Summary for Centralization Alternatives

Spent Nuclear Fuel Site Option	*Phase 1 Approach*	*Phase 1 Port-to-Site Shipments*	*Site-to-Site Shipments*[a]	*Phase 2 or Port-to-Final Site Shipments*	*Total Number of Shipments*
INEL	N/A - Single phase program			837	837
SRS	N/A - Single phase program			837	837
NTS	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
ORR	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	998/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902
Hanford Site	Geographic	East to SRS: 501 West to INEL: 143	From SRS: 126/51 From INEL: 36/15	From East: 150 From West: 43	999/903
	By Fuel	MTR to SRS: 520 TRIGA to INEL: 124	From SRS: 130/52 From INEL: 31/13	From East: 150 From West: 43	998/902
	All SRS	644	161/65	From East: 150 From West: 43	998/902
	All INEL	644	161/65	From East: 150 From West: 43	998/902

[a] *Truck/Rail shipments assuming that the truck casks used for intersite shipments are capable of carrying 4 times as much fuel and rail casks 10 times as much fuel as the shipping cask received from the foreign research reactor due to consolidation.*

INEL = Idaho National Engineering Laboratory; ORR = Oak Ridge Reservation; SRS = Savannah River Site; NTS = Nevada Test Site; HS = Hanford Site

E.4 Truck and Rail Routing Analysis

Both rail and highway shipping capabilities are available at all potential ports of entry, and each of the five DOE sites is or could be made capable of receiving foreign research reactor spent nuclear fuel transported by rail or highway. Therefore, shipment of spent nuclear fuel will be analyzed along representative highway and railway routes for all ports and SNF&INEL Final EIS (DOE, 1995) alternatives.

As discussed above, each alternative can be defined as a set of origin and destination pairs representing shipment linkages between ports of entry and interim management sites. The calculation of the overland transportation risk for an alternative depends in part on characteristics of the transportation routes between the origin and destination sites. Regulatory routing criteria and the methods used to determine representative truck and rail routes for the transportation risk assessment are described below. In addition, the route characteristics that are important for the risk assessment are summarized.

E.4.1 Routing Regulations

Department of Transportation's public highway routing regulations are prescribed in 49 CFR Part 397. The regulations' objectives are to reduce the impacts of transporting radioactive materials, to establish consistent and uniform requirements for route selection, and to identify the role of State and local governments in the routing of radioactive materials. The regulations attempt to reduce potential hazards by avoiding populous areas and by minimizing travel times. Further, they require that the carrier of radioactive materials ensure that the vehicle is operated on routes that minimize radiological risks, and that accident rates, transit times, population density and activity, time of day, and day of week are considered in determining risk.

A shipment of a "highway route controlled quantity" of radioactive material, such as spent nuclear fuel, is required by 49 CFR 397 Subpart D to use the interstate highway system except when moving from origin to interstate or from interstate to destination, when making necessary repair or rest stops, or when emergency conditions make continued use of the interstate unsafe or impossible. Carriers are required to use interstate circumferential or bypass routes, if available, to avoid populous areas. Other "preferred highways" may be designated by any State or Tribe to replace or supplement the interstate system (DOT, 1992). Under its authority to regulate interstate transportation safety, the Department of Transportation can prohibit State and local bans and restrictions as "undue restraint of interstate commerce." State or local bans will be pre-empted if inconsistent with 49 CFR 397.

Currently, there are no Department of Transportation railroad routing regulations specific to the transportation of radioactive materials. Routes are generally fixed by the location of rail lines, and urban areas cannot be readily bypassed.

E.4.2 Determination of Representative Transportation Routes

Representative overland truck and rail routes have been determined for all pairs of origin and destination sites considered by the alternatives. The routes were selected consistent with current routing practices and all applicable routing regulations and guidelines. However, because the routes were determined for risk assessment purposes, they do not necessarily represent the actual routes that would be used to transport foreign research reactor spent nuclear fuel in the future. Specific routes cannot be identified in advance because the route would not be finalized until it had been reviewed and approved by the NRC. The selection of the actual route would be responsive to environmental and other conditions that were in effect or could reasonably be predicted at the time of shipment. Such conditions could include adverse weather conditions, truck or road conditions, bridge closures, etc. (Massey, 1994).

For both truck and rail transportation modes, the route characteristics that are important to the radiological risk assessment include the total shipment distance between each origin and destination pair and the population distribution along the route. The specific route selected determines both the total potentially exposed population and the expected frequency of transportation-related accidents. Route characteristics are summarized in Tables E-3 and E-4 for the ports of entry and management sites considered in this assessment. The ports of Philadelphia, PA; Elizabeth, NJ; and Long Beach, CA are included in the list to show the effects on overland transportation of choosing high population ports. The routes from Canada are representative for risk analysis purposes, many other routes are available for use. They are not included in the risk analysis described later in this appendix. The exposed population includes all persons living within 800 m [0.5 mile (mi)] on each side of the route. The representative routes are shown in Attachment 1 to this appendix.

Table E-3 Summary of Route Distances for Truck and Rail Modes

Shipments to Hanford Site:				
Route[a]	**Distance km (mi)**	**Percentage in Zone**		
		Rural	**Suburban**	**Urban**
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	4579 (2858)	85.5	13.3	1.2
Charleston, SC (Wando Terminal)	4603 (2873)	85.3	13.4	1.3
Elizabeth, NJ	4527 (2812)	84.4	14.1	1.4
Galveston, TX	3746 (2327)	86.0	11.8	2.3
Jacksonville, FL	4708 (2924)	83.9	14.6	1.5
Newport News, VA	4682 (2908)	84.9	13.3	1.8
Norfolk, VA	4748 (2949)	84.4	13.8	1.7
Philadelphia, PA	4617 (2868)	82.9	15.6	1.5
Portsmouth, VA	4717 (2930)	84.5	13.6	1.9
Savannah, GA	4529 (2813)	84.7	13.8	1.5
MOTSU, NC	4617 (2868)	85.7	13.1	1.3
Wilmington, NC	4770 (2963)	85.3	13.5	1.1
Rail:				
Charleston, SC (NWS)	4925 (3059)	84.5	13.7	1.8
Charleston, SC (Wando Terminal)	4925 (3059)	84.5	13.7	1.8
Elizabeth, NJ	4846 (3010)	76.1	19.5	4.4
Galveston, TX	3851 (2392)	89.9	9.1	1.0
Jacksonville, FL	4941 (3069)	85.4	13.0	1.6
Newport News, VA	4972 (3088)	83.6	13.7	2.7
Norfolk, VA	5131 (3187)	83.8	13.6	2.7
Philadelphia, PA	4769 (2962)	77.1	18.6	4.3
Portsmouth, VA	5083 (3157)	84.0	13.4	2.6
Savannah, GA	4977 (3091)	85.3	13.2	1.4
MOTSU, NC	5157 (3203)	83.6	14.8	1.5
Wilmington, NC	5142 (3194)	83.7	14.7	1.5
From Western Ports				
Truck:				
Long Beach CA	1986 (1241)	80.5	14.3	5.2
NWS Concord, CA	1378 (856)	79.4	18.0	2.6
Portland, OR	407 (253)	81.5	15.3	3.3
Tacoma, WA	399 (248)	73.4	22.8	3.8
Rail:				
Long Beach, CA	2553 (1587)	85.6	8.9	5.5
NWS Concord, CA	1531 (951)	80.3	14.7	5.0
Portland, OR	385 (239)	82.1	13.4	4.5
Tacoma, WA	602 (374)	79.2	17.2	3.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	4456 (2768)	82.8	15.6	1.6
Idaho National Engineering Laboratory	964 (599)	91.3	7.6	1.1
Nevada Test Site	1816 (1128)	86.5	10.9	2.6
Oak Ridge Reservation	3967 (2464)	87.8	11.0	1.2
Savannah River	4390 (2727)	84.3	14.2	1.5
Sweetgrass, MT	1407 (874)	89.4	10.0	0.6
Rail:				
Alexandria Bay, NY	4634 (2878)	79.6	16.6	3.8
Idaho National Engineering Laboratory	1059 (658)	91.4	7.1	1.4

Shipments to Hanford Site:				
Route[a]	Distance km (mi)	Percentage in Zone		
		Rural	Suburban	Urban
Nevada Test Site	2096 (1302)	93.0	5.9	1.1
Oak Ridge Reservation	4188 (2601)	91.2	7.4	1.3
Savannah River	4754 (2953)	84.7	13.5	1.8
Sweetgrass, MT	976 (606)	91.7	6.8	1.6

Shipments to Idaho National Engineering Laboratory:				
Route[a]	Distance km (mi)	Percentage in Zone		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	3910 (2441)	84.4	14.3	1.3
Charleston, SC (Wando Terminal)	3935 (2456)	84.2	14.4	1.4
Elizabeth, NJ	3858 (2396)	82.9	15.5	1.5
Galveston, TX	3077 (1911)	84.5	13.0	2.5
Jacksonville, FL	4031 (2504)	82.5	15.9	1.5
Newport News, VA	4012 (2492)	83.5	14.6	1.9
Norfolk, VA	4073 (2530)	83.1	15.1	1.8
Philadelphia, PA	3948 (2452)	81.2	17.2	1.6
Portsmouth, VA	4048 (2514)	83.1	14.8	2.1
Savannah, GA	3861 (2398)	83.3	15.1	1.6
MOTSU, NC	3875 (2407)	85.3	13.5	1.2
Wilmington, NC	4099 (2546)	84.1	14.8	1.2
Rail:				
Charleston, SC (NWS)	4046 (2513)	82.6	15.3	2.1
Charleston, SC (Wando Terminal)	4046 (2513)	82.6	15.3	2.1
Elizabeth, NJ	3967 (2464)	72.3	22.5	5.2
Galveston, TX	2972 (1846)	88.9	10.1	1.0
Jacksonville, FL	4062 (2523)	83.7	14.6	1.7
Newport News, VA	4093 (2542)	81.5	15.4	3.1
Norfolk, VA	4252 (2641)	81.8	15.2	3.0
Philadelphia, PA	3890 (2416)	73.4	21.5	5.1
Portsmouth, VA	4204 (2611)	82.1	14.9	3.0
Savannah, GA	4097 (2545)	83.6	14.8	1.6
MOTSU, NC	4278 (2657)	81.6	16.7	1.7
Wilmington, NC	4263 (2648)	81.8	16.5	1.7
From Western Ports				
Truck:				
Long Beach, CA	1575 (979)	77.7	14.2	8.1
NWS Concord, CA	1518 (943)	85.9	11.1	3.1
Portland, OR	1188 (738)	88.6	9.8	1.7
Tacoma, WA	1312 (815)	87.0	11.4	1.6
Rail:				
Long Beach, CA	1675 (1041)	81.5	10.5	8.0
NWS Concord, CA	1473 (915)	89.0	8.7	2.4
Portland, OR	1264 (785)	92.6	5.8	1.6
Tacoma, WA	1504 (934)	88.6	9.2	2.2
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	3787 (2352)	81.0	17.2	1.7
Hanford Site	964 (599)	91.3	7.6	1.1
Nevada Test Site	1146 (712)	82.8	13.7	3.5

Shipments to Idaho National Engineering Laboratory:				
		Percentage in Zone		
Route[a]	Distance km (mi)	Rural	Suburban	Urban
Oak Ridge Reservation	3297 (2048)	86.8	12.0	1.2
Savannah River	3721 (2311)	82.8	15.6	1.6
Sweetgrass, MT	874 (543)	94.8	4.8	0.4
Rail:				
Alexandria Bay, NY	3755 (2332)	76.4	19.1	4.5
Hanford Site	1059 (658)	91.4	7.1	1.4
Nevada Test Site	1217 (756)	92.8	5.9	1.3
Oak Ridge Reservation	3309 (2055)	90.7	7.8	1.5
Savannah River	3875 (2407)	82.8	15.2	2.0
Sweetgrass, MT	1982 (1231)	93.2	5.8	1.0

Shipments to Nevada Test Site:				
		Percentage in Zone		
Route[a]	Distance km (mi)	Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	3930 (2543)	84.5	14.1	1.4
Charleston, SC (Wando Terminal)	4098 (2558)	84.3	14.2	1.5
Elizabeth, NJ	4302 (2672)	80.5	17.2	2.3
Galveston ,TX	2998 (1862)	85.4	11.5	3.2
Jacksonville, FL	4197 (2607)	82.8	15.4	1.8
Newport News, VA	4178 (2595)	83.8	14.1	2.1
Norfolk, VA	4239 (2633)	83.4	14.6	2.0
Philadelphia, PA	4223 (2623)	80.4	17.4	2.2
Portsmouth, VA	4213 (2617)	83.4	14.3	2.3
Savannah, GA	4027 (2501)	83.6	14.6	1.8
MOTSU, NC	3956 (2457)	83.0	15.0	2.0
Wilmington, NC	4267 (2650)	84.3	14.3	1.4
Rail:				
Charleston, SC (NWS)	4741 (2945)	84.3	13.7	2.0
Charleston, SC (Wando Terminal)	4741 (2945)	84.3	13.7	2.0
Elizabeth, NJ	4661 (2895)	75.6	19.7	4.7
Galveston, TX	3148 (1955)	92.0	7.2	0.8
Jacksonville, FL	4758 (2955)	85.3	13.1	1.7
Newport News, VA	4787 (2973)	83.4	13.8	2.9
Norfolk, VA	4948 (3073)	83.6	13.6	2.8
Philadelphia, PA	4585 (2848)	76.6	18.8	4.6
Portsmouth, VA	4898 (3042)	83.8	13.4	2.8
Savannah, GA	4793 (2977)	85.2	13.2	1.5
MOTSU, NC	4973 (3089)	83.4	14.9	1.7
Wilmington, NC	4959 (3080)	83.5	14.8	1.7
From Western Ports				
Truck:				
Long Beach, CA	645 (401)	71.3	12.7	16.0
NWS Concord, CA	1146 (712)	81.8	11.3	6.9
Portland, OR	2045 (1270)	85.5	11.5	2.9
Tacoma, WA	2164 (1344)	84.7	12.6	2.7
Rail:				
Long Beach, CA	777 (483)	70.5	14.3	15.3
NWS Concord, CA	1369 (850)	77.8	16.7	5.6
Portland, OR	2301 (1429)	93.5	5.3	1.2

Shipments to Nevada Test Site:				
Route[a]	**Distance km (mi)**	***Percentage in Zone***		
		Rural	**Suburban**	**Urban**
Tacoma, WA	2542 (1579)	91.0	7.4	1.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	4217 (2619)	82.0	16.0	1.9
Hanford Site	1816 (1128)	86.5	10.9	2.6
Idaho National Engineering Laboratory	1146 (712)	82.8	13.7	3.5
Oak Ridge Reservation	3463 (2151)	86.9	11.5	1.6
Savannah River	3887 (2414)	83.1	15.1	1.8
Sweetgrass, MT	1900 (1180)	87.5	10.0	2.5
Rail:				
Alexandria Bay, NY	4448 (2763)	79.2	16.7	4.0
Hanford Site	2096 (1302)	93.0	5.9	1.1
Idaho National Engineering Laboratory	1217 (756)	92.8	5.9	1.3
Oak Ridge Reservation	4004 (2487)	91.4	7.2	1.5
Savannah River	4571 (2839)	84.5	13.5	1.9
Sweetgrass, MT	3019 (1875)	93.7	5.4	0.9

Shipments to Oak Ridge Reservation:				
Route[a]	**Distance km (mi)**	***Percentage in Zone***		
		Rural	**Suburban**	**Urban**
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	644 (402)	71.6	27.6	0.8
Charleston, SC (Wando Terminal)	668 (417)	70.9	27.8	1.3
Elizabeth, NJ	1188 (738)	62.2	35.7	2.1
Galveston, TX	1550 (963)	73.3	24.6	2.1
Jacksonville, FL	913 (567)	66.8	32.0	1.3
Newport News, VA	890 (553)	69.8	27.6	2.6
Norfolk, VA	886 (550)	68.4	30.2	1.3
Philadelphia, PA	1095 (680)	64.7	31.7	3.6
Portsmouth, VA	926 (575)	68.4	28.2	3.4
Savannah, GA	723 (449)	74.3	25.0	0.6
MOTSU, NC	799 (496)	72.4	26.7	0.9
Wilmington, NC	819 (509)	72.6	26.5	0.9
Rail:				
Charleston, SC (NWS)	935 (581)	65.2	33.3	1.5
Charleston, SC (Wando Terminal)	935 (581)	65.2	33.3	1.5
Elizabeth, NJ	1264 (785)	44.7	43.2	12.2
Galveston, TX	1695 (1053)	70.5	26.2	3.3
Jacksonville, FL	910 (565)	65.7	31.9	2.4
Newport News, VA	1230 (764)	59.2	38.7	2.0
Norfolk, VA	1109 (689)	62.2	36.3	1.6
Philadelphia, PA	1129 (701)	48.6	43.0	8.4
Portsmouth, VA	1061 (659)	62.3	36.4	1.3
Savannah, GA	945 (587)	66.2	32.1	1.7
MOTSU, NC	873 (542)	61.5	37.1	1.5
Wilmington, NC	857 (532)	61.7	36.8	1.5
From Western Ports				
Truck:				
Long Beach, CA	3614 (2246)	85.0	11.0	3.8
NWS Concord, CA	4117 (2557)	86.3	10.9	2.8

Shipments to Oak Ridge Reservation:				
Route[a]	Distance km (mi)	Percentage in Zone		
		Rural	Suburban	Urban
Portland, OR	4200 (2609)	87.0	11.5	1.5
Tacoma, WA	4279 (2658)	88.0	11.0	1.0
Rail:				
Long Beach, CA	4302 (2674)	86.5	9.7	3.9
NWS Concord, CA	4524 (2810)	87.5	10.4	2.2
Portland, OR	4551 (2827)	85.5	12.1	2.4
Tacoma, WA	4568 (2837)	83.7	13.3	3.0
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	1492 (927)	65.9	33.5	0.7
Hanford Site	3967 (2464)	87.8	11.0	1.2
Idaho National Engineering Laboratory	3297 (2048)	86.8	12.0	1.2
Nevada Test Site	3463 (2151)	86.9	11.5	1.6
Savannah River	610 (379)	59.1	38.5	2.4
Sweetgrass, MT	1900 (1180)	87.5	10.0	2.5
Rail:				
Alexandria Bay, NY	1565 (972)	57.5	35.7	6.8
Hanford Site	4188 (2601)	91.2	7.4	1.3
Idaho National Engineering Laboratory	3309 (2055)	90.7	7.8	1.5
Nevada Test Site	4004 (2487)	91.4	7.2	1.5
Savannah River	671 (417)	68.8	29.8	1.4
Sweetgrass, MT	3375 (2096)	83.7	13.7	2.6

Shipments to Savannah River Site:				
Route[a]	Distance km (mi)	Percentage in Zone		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	301 (188)	72.9	26.2	0.9
Charleston, SC (Wando Terminal)	325 (203)	71.6	26.6	1.8
Elizabeth, NJ	1325 (823)	63.8	34.2	2.1
Galveston, TX	1610 (1000)	70.5	27.0	2.5
Jacksonville, FL	607 (377)	81.5	18.4	0.0
Newport News, VA	836 (519)	71.1	26.8	2.1
Norfolk, VA	802 (498)	72.8	26.2	1.0
Philadelphia, PA	1193 (741)	62.1	34.0	3.9
Portsmouth, VA	807 (501)	72.7	26.1	1.1
Savannah, GA	403 (250)	79.1	20.8	0.0
MOTSU, NC	403 (250)	82.5	17.2	0.3
Wilmington, NC	499 (310)	75.5	24.0	0.5
Rail:				
Charleston, SC (NWS)	225 (140)	83.9	13.6	2.5
Charleston, SC (Wando Terminal)	225 (140)	83.9	13.6	2.5
Elizabeth, NJ	1404 (872)	56.2	33.0	10.8
Galveston, TX	1890 (1174)	69.6	26.2	4.2
Jacksonville, FL	417 (259)	83.7	13.7	2.6
Newport News, VA	972 (604)	69.1	28.7	2.2
Norfolk, VA	852 (529)	74.3	24.1	1.6
Philadelphia, PA	1270 (789)	60.9	31.8	7.2
Portsmouth, VA	803 (499)	75.2	23.5	1.3
Savannah, GA	184 (114)	87.9	10.9	1.2

Shipments to Savannah River Site:				
Route[a]	*Distance km (mi)*	*Percentage in Zone*		
		Rural	*Suburban*	*Urban*
MOTSU, NC	615 (382)	77.9	20.5	1.6
Wilmington, NC	601 (373)	78.7	19.7	1.6
From Western Ports				
Truck:				
Long Beach, CA	3931 (2443)	78.8	18.0	3.3
NWS Concord, CA	4482 (2784)	79.4	17.2	3.3
Portland, OR	4635 (2879)	83.9	14.4	1.7
Tacoma, WA	4719 (2931)	84.8	13.9	1.3
Rail:				
Long Beach, CA	5212 (3239)	80.9	15.3	3.7
NWS Concord, CA	5123 (3182)	80.0	16.4	3.6
Portland, OR	5078 (3154)	82.0	15.4	2.6
Tacoma, WA	5096 (3165)	80.4	16.5	3.1
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	1629 (1012)	66.8	32.4	0.8
Hanford Site	4390 (2727)	84.3	14.2	1.5
Idaho National Engineering Laboratory	3721 (2311)	82.8	15.6	1.6
Nevada Test Site	3887 (2414)	83.1	15.1	1.8
Oak Ridge Reservation	610 (379)	59.1	38.5	2.4
Sweetgrass, MT	4147 (2576)	85.2	13.6	1.3
Rail:				
Alexandria Bay, NY	2062 (1281)	53.8	35.5	10.7
Hanford Site	4754 (2953)	84.7	13.5	1.8
Idaho National Engineering Laboratory	3875 (2407)	82.8	15.2	2.0
Nevada Test Site	4571 (2839)	84.5	13.5	1.9
Oak Ridge Reservation	671 (417)	68.8	29.8	1.4
Sweetgrass, MT	3903 (2424)	79.4	17.8	2.8

[a] *Route characteristics were generated using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b) for truck and rail modes, respectively.*

Table E-4 Summary of the Population Distributions Along Routes for Truck and Rail Modes

Shipments to Hanford Site:				
Route[a]	*Number of Affected Persons*[b]	*Average Persons/km*^{2}		
		Rural	*Suburban*	*Urban*
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	550,000	7.0	342.5	2149.1
Charleston, SC (Wando Terminal)	569,000	7.0	346.1	2158.6
Elizabeth, NJ	585,000	7.8	318.3	2233.1
Galveston, TX	575,000	4.9	401.5	2139.5
Jacksonville, FL	643,000	7.1	338.6	2180.5
Newport News, VA	677,000	7.5	356.9	2254.3
Norfolk, VA	694,000	7.6	362.0	2219.3
Philadelphia, PA	622,000	7.4	317.4	2079.3
Portsmouth, VA	718,000	7.5	364.3	2243.7
Savannah, GA	602,000	6.8	344.2	2205.1
MOTSU, NC	548,000	7.6	332.1	2146.8

Shipments to Hanford Site:				
Route[a]	**Number of Affected Persons[b]**	**Average Persons/km²**		
		Rural	**Suburban**	**Urban**
Wilmington, NC	556,000	7.5	330.0	2149.9
Rail:				
Charleston, SC (NWS)	731,000	6.9	354.6	2296.8
Charleston, SC (Wando Terminal)	731,000	6.9	354.6	2296.8
Elizabeth, NJ	1,380,000	7.2	355.0	2506.4
Galveston, TX	347,000	4.8	374.8	2034.6
Jacksonville, FL	657,000	6.9	343.6	2272.5
Newport News, VA	936,000	7.5	329.1	2623.1
Norfolk, VA	960,000	7.6	338.8	2592.7
Philadelphia, PA	1,350,000	7.1	358.5	2567.1
Portsmouth, VA	934,000	7.6	334.2	2608.4
Savannah, GA	641,000	7.0	343.1	2244.5
MOTSU, NC	739,000	7.7	346.1	2288.1
Wilmington, NC	736,000	7.7	346.7	2288.1
From Western Ports				
Truck:				
Long Beach, CA	617,000	7.9	381.0	2693.6
NWS Concord, CA	263,000	9.3	335.1	2159.0
Portland, OR	85,700	6.3	413.3	2088.6
Tacoma, WA	98,600	7.7	321.9	2120.5
Rail:				
Long Beach, CA	783,000	3.6	471.4	2781.1
NWS Concord, CA	419,000	7.0	368.7	2363.7
Portland, OR	99,500	6.1	450.0	2294.4
Tacoma, WA	136,000	10.6	355.9	2161.1
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	612,000	7.7	300.4	2211.8
Idaho National Engineering Laboratory	82,800	5.5	363.0	2034.6
Nevada Test Site	305,000	4.1	447.3	2176.8
Oak Ridge Reservation	429,000	6.0	351.1	2207.3
Savannah River	599,000	6.7	354.7	2198.1
Sweetgrass, MT	106,000	4.5	314.4	2152.3
Rail:				
Alexandria Bay, NY	1,170,000	7.0	360.2	2584.5
Idaho National Engineering Laboratory	95,400	4.2	373.6	1935.8
Nevada Test Site	157,000	3.5	402.3	1980.5
Oak Ridge Reservation	410,000	6.7	375.7	2220.3
Savannah River	690,000	6.8	355.8	2267.6
Sweetgrass, MT	92,400	4.1	394.4	1979.9

Shipments to Idaho National Engineering Laboratory:				
Route[a]	**Number of Affected Persons[b]**	**Average Persons/km²**		
		Rural	**Suburban**	**Urban**
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	498,000	7.4	334.0	2157.4
Charleston, SC (Wando Terminal)	518,000	7.4	338.1	2167.2
Elizabeth, NJ	536,000	8.5	315.6	2257.0
Galveston, TX	526,000	5.1	405.8	2149.1

Shipments to Idaho National Engineering Laboratory:				
Route[a]	***Number of Affected Persons[b]***	***Average Persons/km^2***		
		Rural	***Suburban***	***Urban***
Jacksonville, FL	576,000	7.6	332.2	2224.8
Newport News, VA	628,000	8.0	356.8	2274.7
Norfolk, VA	631,000	8.1	362.5	2220.6
Philadelphia, PA	573,000	7.9	314.9	2084.2
Portsmouth, VA	670,000	8.0	364.7	2261.3
Savannah, GA	553,000	7.3	343.2	2224.8
MOTSU, NC	463,000	8.1	327.9	2155.2
Wilmington, NC	507,000	8.1	328.1	2166.9
Rail:				
Charleston, SC (NWS)	671,000	7.6	348.5	2332.6
Charleston, SC (Wando Terminal)	671,000	7.6	348.5	2332.6
Elizabeth, NJ	1, 320,000	8.2	350.8	2528.3
Galveston, TX	286,000	5.1	365.7	2068.7
Jacksonville, FL	597,000	7.7	336.3	2312.5
Newport News, VA	875,000	8.4	321.1	2665.7
Norfolk, VA	900,000	8.5	331.7	2632.7
Philadelphia, PA	1,290,000	8.1	354.2	2592.0
Portsmouth, VA	874,000	8.6	326.6	2650.6
Savannah, GA	580,000	7.8	335.9	2284.9
MOTSU, NC	679,000	8.7	340.2	2328.4
Wilmington, NC	675,000	8.7	340.8	2328.4
From Western Ports				
Truck:				
Long Beach, CA	692,000	3.8	487.0	2641.1
NWS Concord, CA	271,000	3.5	411.6	2181.5
Portland, OR	143,000	5.6	395.0	2082.7
Tacoma, WA	157,000	6.1	336.5	2098.8
Rail:				
Long Beach, CA	722,000	3.5	484.6	2830.1
NWS Concord, CA	198,000	4.4	337.2	2293.0
Portland, OR	116,000	4.3	330.2	2222.6
Tacoma, WA	199,000	6.1	326.5	2291.5
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	564,000	8.3	296.8	2230.8
Hanford Site	82,800	5.5	363.0	2034.6
Nevada Test Site	256,000	3.9	470.3	2201.5
Oak Ridge Reservation	380,000	6.3	350.4	2237.4
Savannah River	551,000	7.2	354.4	2217.9
Sweetgrass, MT	38,900	4.3	348.1	2057.3
Rail:				
Alexandria Bay NY	1,110,000	7.9	355.3	2614.7
Hanford Site	95,400	4.2	373.6	1935.8
Nevada Test Site	96,100	3.3	384.6	2022.2
Oak Ridge Reservation	350,000	7.5	365.3	2270.2
Savannah River	630,000	7.6	349.6	2303.3
Sweetgrass, MT	134,000	4.2	338.5	2068.1

Shipments to Nevada Test Site:				
Route[a]	Number of Affected Persons[b]	Average Persons/km^2		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	540,000	6.7	347.4	2179.0
Charleston, SC (Wando Terminal)	559,000	6.7	351.2	2186.6
Elizabeth, NJ	782,000	7.5	343.2	2300.4
Galveston, TX	595,000	4.1	463.8	2277.2
Jacksonville, FL	639,000	6.9	344.9	2265.9
Newport News, VA	691,000	7.3	370.6	2301.4
Norfolk, VA	694,000	7.4	375.5	2257.2
Philadelphia, PA	756,000	7.6	349.2	2199.9
Portsmouth, VA	732,000	7.3	378.0	2287.8
Savannah, GA	616,000	6.5	357.1	2265.9
MOTSU, NC	619,000	8.6	336.2	2218.3
Wilmington, NC	570,000	7.4	341.6	2229.9
Rail:				
Charleston, SC (NWS)	733,000	6.5	362.0	2314.6
Charleston, SC (Wando Terminal)	733,000	6.5	362.0	2314.6
Elizabeth, NJ	1,390,000	6.8	360.2	2511.5
Galveston, TX	231,000	4.3	374.6	2124.4
Jacksonville, FL	659,000	6.6	350.9	2294.0
Newport News, VA	938,000	7.1	335.5	2629.1
Norfolk, VA	963,000	7.3	345.5	2599.2
Philadelphia, PA	1,350,000	6.7	364.2	2571.7
Portsmouth, VA	936,000	7.3	340.8	2614.8
Savannah, GA	643,000	6.7	350.2	2268.7
MOTSU, NC	742,000	7.4	352.3	2308.6
Wilmington, NC	738,000	7.4	353.0	2308.6
From Western Ports				
Truck:				
Long Beach, CA	518,000	2.6	550.5	2817.2
NWS Concord, CA	437,000	3.8	559.2	2617.4
Portland, OR	375,000	4.3	452.0	2154.7
Tacoma, WA	379,000	4.7	409.3	2174.8
Rail:				
Long Beach, CA	628,000	3.4	522.1	2934.2
NWS Concord, CA	407,000	6.2	360.9	2313.2
Portland, OR	177,000	3.6	376.6	2183.0
Tacoma, WA	261,000	005	357.6	2251.6
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay NY	644,000	7.5	308.8	2262.5
Hanford Site	305,000	4.1	447.3	2176.8
Idaho National Engineering Laboratory	256,000	3.9	470.3	2201.5
Oak Ridge Reservation	443,000	5.5	370.8	2291.1
Savannah River	613,000	6.4	368.3	2261.8
Sweetgrass, MT	304,000	4.0	455.9	2167.2
Rail:				
Alexandria Bay NY	1,170,000	6.6	366.8	2589.6
Hanford Site	157,000	3.5	402.3	1980.5

Shipments to Nevada Test Site:				
Route[a]	**Number of Affected Persons[b]**	**Average Persons/km^2**		
		Rural	**Suburban**	**Urban**
Idaho National Engineering Laboratory	96,100	3.3	384.6	2022.2
Oak Ridge Reservation	413,000	6.3	394.1	2252.1
Savannah River	692,000	6.4	363.7	2287.5
Sweetgrass, MT	196,000	3.7	370.8	2067.5

Shipments to Oak Ridge Reservation:				
Route[a]	**Number of Affected Persons[b]**	**Average Persons/km^2**		
		Rural	**Suburban**	**Urban**
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	108,000	15.0	297.5	1842.7
Charleston, SC (Wando Terminal)	127,000	14.7	311.5	2027.0
Elizabeth, NJ	290,000	19.9	273.2	2343.1
Galveston, TX	337,000	13.5	330.3	2358.5
Jacksonville, FL	175,000	15.3	266.0	2322.6
Newport News, VA	209,000	18.5	286.8	2316.6
Norfolk, VA	174,000	17.6	292.1	2073.3
Philadelphia, PA	335,000	18.8	335.1	2215.7
Portsmouth, VA	251,000	18.2	309.4	2270.2
Savannah, GA	101,000	14.1	274.1	1764.7
MOTSU, NC	128,000	17.6	283.4	1854.5
Wilmington, NC	128,000	16.7	280.4	1764.7
Rail:				
Charleston, SC (NWS)	194,000	17.4	272.6	2202.7
Charleston, SC (Wando Terminal)	194,000	17.4	272.6	2202.7
Elizabeth, NJ	949,000	17.1	353.3	2694.7
Galveston, TX	471,000	13.4	360.1	2306.2
Jacksonville, FL	235,000	11.6	335.9	2233.3
Newport News, VA	305,000	16.8	277.3	2175.5
Norfolk, VA	241,000	17.3	270.6	2077.6
Philadelphia, PA	649,000	16.9	333.2	2653.3
Portsmouth, VA	215,000	17.8	259.8	2075.0
Savannah, GA	218,000	12.1	335.1	2090.0
MOTSU, NC	186,000	17.7	263.7	2038.3
Wilmington, NC	183,000	17.8	263.3	2038.3
From Western Ports				
Truck:				
Long Beach, CA	823,000	6.2	380.8	2607.0
NWS Concord, CA	742,000	6.0	401.1	2426.5
Portland, OR	519,000	6.0	367.2	2195.4
Tacoma, WA	431,000	6.2	343.7	2213.8
Rail:				
Long Beach, CA	995,000	6.6	398.5	2736.4
NWS Concord, CA	664,000	6.9	379.3	2317.2
Portland, OR	765,000	7.6	393.4	2272.1
Tacoma, WA	919,000	7.7	407.7	2330.2
From DOE Sites/Canadian Border				
Truck:				
Alexandria Bay, NY	257,000	19.9	258.2	1896.7
Hanford Site	429,000	6.0	351.0	2207.3

Shipments to Oak Ridge Reservation:				
Route[a]	Number of Affected Persons[b]	Average Persons/km^2		
		Rural	Suburban	Urban
Idaho National Engineering Laboratory	380,000	6.3	350.0	2237.4
Nevada Test Site	443,000	5.5	371.0	2291.1
Savannah River	175,000	17	318.0	2244.1
Sweetgrass, MT	346,000	6.3	336.3	2180.9
Rail:				
Alexandria Bay NY	752,000	18.2	378.0	2443.0
Hanford Site	416,000	6.7	376.0	2220.3
Idaho National Engineering Laboratory	350,000	7.5	365.0	2270.2
Nevada Test Site	413,000	6.3	394.0	2252.1
Savannah River	132,000	15.2	289.0	2164.4
Sweetgrass, MT	627,000	8.7	395.5	2256.5

Shipments to Savannah River Site:				
Route[a]	Number of Affected Persons[b]	Average Person/km^2		
		Rural	Suburban	Urban
From Eastern Ports				
Truck:				
Charleston, SC (NWS)	46,200	16.3	275.0	1764.7
Charleston, SC (Wando Terminal)	65,700	15.6	306.1	2077.9
Elizabeth, NJ	316,000	17.6	277.6	2377.5
Galveston, TX	430,000	12.7	359.1	2254.1
Jacksonville, FL	46,900	13.2	211.4	1764.7
Newport News, VA	181,000	16.2	302.9	2281.5
Norfolk, VA	131,000	16.4	283.9	2007.9
Philadelphia, PA	397,000	16.5	348.5	2228.9
Portsmouth, VA	135,000	16.3	281.8	2033.1
Savannah, GA	37,300	13.6	233.4	1764.7
MOTSU, NC	34,200	15.0	213.0	1925.6
Wilmington, NC	64,700	17.7	256.7	1764.7
Rail:				
Charleston, SC (NWS)	41,200	6.8	328.6	2735.5
Charleston, SC (Wando Terminal)	41,200	6.8	328.6	2735.5
Elizabeth, NJ	903,000	14.2	353.0	2726.4
Galveston, TX	609,000	11.9	394.0	2330.3
Jacksonville, FL	72,200	10.6	290.3	2466.3
Newport News, VA	218,000	13.2	285.6	2444.8
Norfolk, VA	153,000	13.5	275.3	2469.8
Philadelphia, PA	603,000	14.0	328.8	2704.1
Portsmouth, VA	128,000	13.9	253.7	2615.8
Savannah, GA	21,300	9.6	309.0	2707.8
MOTSU, NC	99,000	12.7	260.9	2580.4
Wilmington, NC	95,500	12.8	259.6	2580.4
From Western Ports				
Truck:				
Long Beach, CA	714,000	7.5	369.4	2905.8
NWS Concord, CA	1,040,000	7.3	378.5	2381.7
Portland, OR	686,000	6.7	365.2	2188.9
Tacoma, WA	601,000	6.8	349.2	2202.0
Rail:				
Long Beach, CA	1,280,000	6.9	359.9	2653.0

Shipments to Savannah River Site:				
Route[a]	**Number of Affected Persons[b]**	**Average Person/km ^{2}**		
		Rural	**Suburban**	**Urban**
NWS Concord, CA	1,210,000	7.1	381.6	2369.0
Portland, OR	950,000	7.5	369.0	2246.2
Tacoma, WA	1,100,000	7.5	381.0	2300.7
From DOE Sites/Canadian Border				
Truck:				
Alexandria, Bay NY	284,000	18.0	262.7	2072.4
Hanford Site	599,000	6.7	354.7	2198.1
Idaho National Engineering Laboratory	551,000	7.2	354.4	2217.9
Nevada Test Site	613,000	6.4	368.3	2261.8
Oak Ridge Reservation	175,000	17.0	317.7	2244.1
Sweetgrass, MT	513,000	7.1	344.0	2175.8
Rail:				
Alexandria Bay, NY	1,340,000	14.8	333.1	2756.8
Hanford Site	690,000	6.8	355.8	2267.6
Idaho National Engineering Laboratory	630,000	7.6	349.6	2303.3
Nevada Test Site	692,000	6.4	363.7	2287.5
Oak Ridge Reservation	132,000	15.2	289.2	2164.4
Sweetgrass, MT	812,000	8.4	367.3	2228.5

[a] *Route characteristics were generated using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b) for truck and rail modes, respectively.*

[b] *The affected population includes all persons within 800 m (0.5 mi) of the route.*

The representative truck and rail routes were determined by using the routing models HIGHWAY (Johnson et al., 1993a) and INTERLINE (Johnson et al., 1993b), respectively. These models are described briefly below.

The HIGHWAY computer program is used for selecting highway routes for transporting radioactive materials within the United States by truck. The HIGHWAY data base is a computerized road atlas that currently describes approximately 386,400 kilometer (km) (240,000 mi) of roads. A complete description of the Interstate System and all United States highways is included in the database. In addition, most of the principal State highways and a number of local and community highways are also identified. The code is updated periodically to reflect current road conditions and has been benchmarked against reported mileages and observations of commercial truck firms.

Routes are calculated within the model by minimizing the total impedance between the origin and the destination. The impedance is basically defined as a function of distance and driving time along a particular highway segment. One of the special features of the HIGHWAY model is its ability to calculate routes that maximize the use of interstate highways. This feature allows the user to select routes for shipment of radioactive materials that conform to Department of Transportation regulations, specifically 49 CFR 397 Subpart D. The population densities along a route are derived from 1990 U.S. Bureau of the Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per km ^{2} (0 to 139 persons per mi ^{2}); the suburban range is 55 to 1,284 persons per km ^{2} (140 to 3,326 persons per mi ^{2}); and urban is taken as all population densities greater than 1,284 persons per km ^{2} (3,326 persons per mi ^{2}).

The INTERLINE computer program is designed to simulate routing of the United States rail system. The INTERLINE database consists of 94 separate subnetworks and represents various competing rail companies in the United States. The database used by INTERLINE was originally based on Federal Railroad Administration data and reflected the United States railroad system in 1974. The data base has since been expanded and modified over the past 2 decades. The code is updated periodically to reflect current track conditions and has been benchmarked against reported mileages and observations of commercial rail firms.

The INTERLINE model uses a shortest-route algorithm that finds the minimum impedance path within an individual subnetwork. A separate routine is used to find paths along the subnetworks. The routes selected for this study used the standard assumptions in the INTERLINE model that simulate the selection process that railroads would use to direct shipments of spent nuclear fuel. The population densities along a route are derived from 1990 U.S. Bureau of the Census data. Rural, suburban, and urban areas are characterized according to the following breakdown: rural population densities range from 0 to 54 persons per km^2 (0 to 139 persons per mi^2); the suburban range is 55 to 1,284 persons per km^2 (140 to 3,326 persons per mi^2); and urban is taken as all population densities greater than 1,284 persons per km^2 (3,326 persons per mi^2).

E.5 Methods for Calculating Transportation Risks

The overland transportation risk assessment approach is summarized in Figure E-13. The first step in the ground transportation analysis was to determine the incident-free and accident risk factors, on a per-shipment basis, for transportation of the various types of spent nuclear fuel. Risk factors, as any risk estimate, are the product of the probability of exposure and the magnitude of the exposure. Accident risk factors were calculated for radiological and nonradiological traffic accidents. The probabilities, which are much lower than one, and the magnitudes of exposure were multiplied, yielding very low risk numbers. Incident-free risk factors were calculated for crew and public exposure to radiation emanating from the cask and public exposure to the chemical toxicity of the transportation vehicle exhaust. The probability of incident-free exposure is unity (one).

Radiological risk factors are expressed in units of rem. Later in the analysis, they will be multiplied by International Commission on Radiation Protection Publication 60 (ICRP, 1991) conversion factors and estimated numbers of shipments (see Section E.7.1) to give risk estimates in units of LCFs. The vehicle emission risk factors are calculated in latent mortalities, and the vehicle accident risk factors are calculated in mortalities. The nonradiological risk factors will be multiplied by the number of shipments.

For each alternative, risks were assessed for both incident-free transportation and accident conditions. For the incident-free assessment, risks were calculated for both collective populations of potentially exposed individuals and for MEIs. The accident assessment consists of two components: 1) a probabilistic accident risk assessment that considers the probabilities and consequences of a range of possible transportation accident environments, including low-probability accidents that have high consequences and high-probability accidents that have low consequences; and 2) an accident consequence assessment that considers only the consequences of the most severe transportation accidents postulated.

The RADTRAN 4 computer code (Neuhauser and Kanipe, 1993) is used for the incident-free and accident risk assessments to estimate the impacts to collective populations. RADTRAN 4 was developed by Sandia National Laboratories to calculate population risk associated with the transportation of radioactive materials by a variety of modes, including truck, rail, air, ship, and barge.

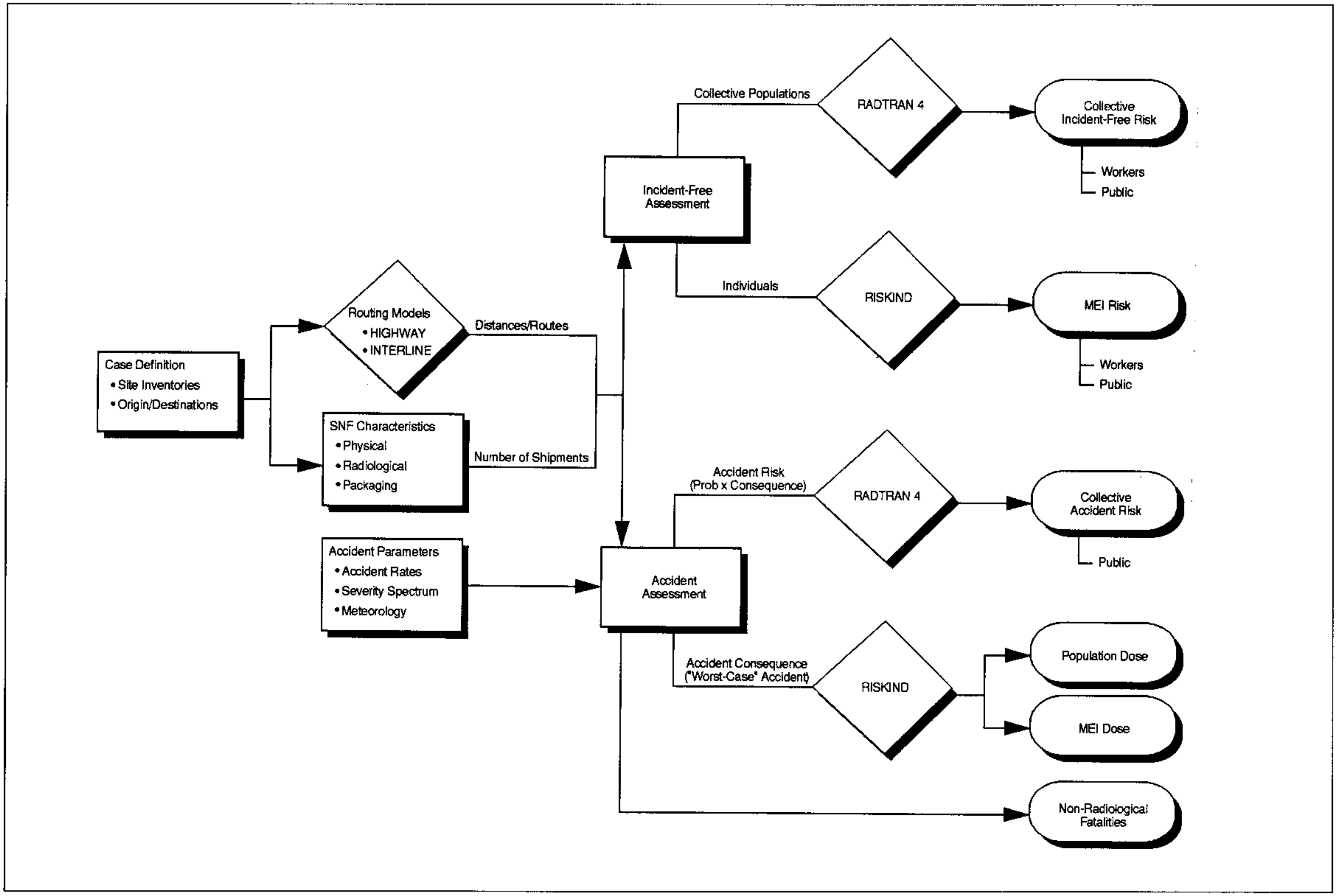


Figure E-13 Summary of the Assessment Approach for the Overland Transportation Risk Assessment

The RADTRAN 4 population risk calculations take into account both the consequences and probabilities of potential exposure events. The collective population risk is a measure of the total radiological risk posed to society as a whole by the alternative being considered. As such, the collective population risk is used as the primary means of comparing the various alternatives.

The RISKIND computer code (Yuan et al., 1993) is used to estimate the incident-free doses to MEIs and for estimating impacts for the accident consequence assessment. The RISKIND computer code was developed for DOE's Office of Civilian Radioactive Waste Management to analyze the exposure of individuals during the incident-free transportation of spent nuclear fuel. In addition, the RISKIND code was designed to allow a detailed assessment of the consequences to individuals and population subgroups from severe spent nuclear fuel transportation accidents under various environmental settings.

The RISKIND calculations were conducted to supplement the collective risk results calculated with RADTRAN 4. Whereas the collective risk results provide a measure of the overall risks of each alternative, the RISKIND calculations are meant to address areas of specific concern to individuals and population subgroups. Essentially, the RISKIND analyses are meant to address "What if" questions, such as, "What if I live next to a site access road?" or "What if an accident happens near my town?"

E.5.1 Incident-Free Risk Assessment Methodology

Radiological dose during normal, incident-free transportation of spent nuclear fuel results from exposure to the external radiation field that surrounds the shipping containers. The dose is a function of the number of people exposed, their proximity to the containers, their length of time of exposure, and the intensity of the radiation field surrounding the containers.

Collective Population Risk: The consequences (dose) during incident-free conditions are expected to occur, therefore, the probability of incident-free consequences is taken to be unity (one) in the RADTRAN 4 code. The radiological risk associated with incident-free transportation conditions results from the potential exposure of people to external radiation in the vicinity of loaded shipments. The maximum allowable external dose rates for exclusive-use shipments were presented in Section E.3.

For incident-free transportation conditions, the RADTRAN 4 computer code considers all major groups of potentially exposed persons. The RADTRAN 4 risk calculations for incident-free highway and rail transportation include exposures of the following population groups:

- Persons along the route (off-link population): Collective doses are calculated for all persons living or working within 800 m (0.5 mi) on each side of a transportation route. The total number of persons within the 1.6 km (1 mi) corridor is calculated separately for each route considered in the assessment.

- Persons sharing the route (on-link population): Collective doses are calculated for persons in all vehicles sharing the transportation route. This group would include persons traveling in the same or opposite direction as the shipment, as well as persons in vehicles passing the shipment.

- Persons at stops: Collective doses are calculated for people who could be exposed while a shipment was stopped en route. For truck transportation, this would include refueling stops, food stops, and rest stops. For rail transportation, stops are assumed to occur for classification purposes.

- Crew Members: Collective doses are calculated for truck and rail transportation crew members.

The doses calculated for the first three population groups are added together to yield the collective dose to the general public. The dose calculated for the fourth group represents the collective dose to workers. The RADTRAN 4 incident-free dose models are not intended to be used for estimating specific risks to individuals.

The RADTRAN 4 incident-free dose calculations are based on expressing the dose rate as a function of distance from a point source (Neuhauser and Kanipe, 1993). Associated with the calculation of incident-free doses for each exposed population group are parameters such as the radiation field strength, source-receptor distance, exposure time, vehicle speed, stop time, traffic density, and route characteristics such as population density. The RADTRAN 4 code user's manual contains derivations of the equations and descriptions of these parameters (Neuhauser and Kanipe, 1993). The values for many of the most important parameters are presented in Section E.6.

The collective incident-free risks are calculated for each specific alternative as follows. Each alternative is first defined as a set of origin and destination pairs. Representative highway and rail routes are determined for each unique pair as described in Section E.4. For each pair, RADTRAN 4 is used to calculate the collective risks to workers and the public for a single shipment based on representative radiological and physical properties of the spent nuclear fuel. These estimates for a single shipment are referred to as per-shipment risk factors. The number of shipments transported across each linkage is then determined for both truck and rail modes. The collective risks for an alternative are calculated by multiplying the number of shipments by the appropriate per-shipment risk factor.

MEI Risk: In addition to the incident-free collective population risk assessment, the risk to MEIs has been estimated for a number of hypothetical exposure events using RISKIND. The receptors include transportation crew members, inspectors, and members of the public exposed during traffic delays, while working at a service station, or living near a port of entry or DOE site.

The dose to each MEI considered is calculated with RISKIND for a given distance, duration, and frequency of exposure specific to that receptor. The distances and durations of exposure are similar to those given in previous transportation assessments and are presented in Section E.6. The exposure scenarios are not meant to be exhaustive, but were selected to provide a realistic range of potential exposure situations.

The RISKIND external dose model considers direct external exposure and exposure from radiation scattered from the ground and air. RISKIND is used to calculate the dose as a function of distance (mrem per hr for stationary exposures and mrem per event for moving shipments) from a spent nuclear fuel shipment based on the dimensions of the shipment. The code models the shipment as a cylindrical volume source; and the calculated dose includes contributions from buildup, cloudshine, and groundshine. The dose rates calculated by using RISKIND have been compared with output from existing shielding codes. The RISKIND code has been found to produce realistic but conservative results. As a conservative measure, potential shielding between the cask and the receptor is not considered.

Nonradiological Risk (Vehicle Related): Vehicle-related health risks resulting from incident-free transport may be associated with the generation of air pollutants by transport vehicles during spent nuclear fuel shipment, and are independent of the radioactive nature of the shipment. The health end point assessed under incident-free transport conditions is the excess latent mortality due to inhalation of vehicle exhaust emissions. Risk factors for pollutant inhalation in terms of latent mortality have been generated (Rao et al., 1982). These risks are $1x10^{-7}$ mortality per km ($1.6x10^{-7}$ per mi) and $1.3x10^{-7}$ mortality per km ($2.1x10^{-7}$ per mi) of truck and rail travel in urban areas, respectively. The risk factors are based on regression analyses of the effects of sulfur dioxide and particulate releases from diesel exhaust on mortality rates. Excess latent mortalities are assumed to be equivalent to LCF. Vehicle-related risks from incident-free transportation are calculated for each case by multiplying the total distance traveled in urban areas by the appropriate risk factor. Similar data are not available for rural and suburban areas.

Risks are summed over the entire route and over all shipments for each spent nuclear fuel case. This method has been used in several reports to calculate risks from incident-free transport. Lack of information for rural and suburban areas is an obvious data gap, although the risk factor would presumably be lower than for urban areas because of lower total emissions from all sources and lower population densities in rural and suburban areas.

E.5.2 Accident Assessment Methodology

The offsite spent nuclear fuel transportation accident analysis considers the impacts of accidents during the transportation of spent nuclear fuel by truck or rail. Under accident conditions, impacts to human health and the environment could result from the release and dispersal of radioactive material. Because of the rigorous design specifications for spent nuclear fuel shipping casks, the NRC has estimated that casks will withstand 99.4 percent of truck or rail accidents without sustaining damage sufficient to breach the cask (Fischer et al., 1987). The 0.6 percent of accidents that could potentially breach the cask are represented by a spectrum of accident severities and radioactive material release conditions. Accident analysis methodology has been developed by the NRC for calculating the probabilities and consequences from this spectrum of unlikely accidents, but it is not possible to predict where along the shipping route such accidents might occur. To provide an assessment of spent nuclear fuel transportation accident impacts, two types of analyses were performed. First, an accident risk assessment was performed that takes into account the probabilities and consequences of a spectrum of accident severities using methodology developed by the NRC (Fischer et al., 1987). The accident risk assessment used route-specific information for accident rates and population densities. For the spectrum of accidents considered in the analysis, accident consequences in terms of collective dose to the population within 80 km (50 mi) were multiplied by the accident probabilities to yield dose risk. Second, to represent the maximum reasonably foreseeable impacts to individuals and populations should an accident occur, radiological consequences were calculated for an accident of maximum credible severity in each population zone. An accident is considered credible if its probability of occurrence is greater than 1×10^{-7} per yr.

Accident Risk Assessment: The risk analysis of potential accidents differs from the incident-free analysis because accident occurrences are statistical in nature. The accident risk assessment is treated probabilistically in RADTRAN 4. Accident risk is defined as the product of the accident consequence (dose) and the probability of the accident occurring. In this respect, the RADTRAN 4 code estimates the collective accident risk to populations by considering a spectrum of transportation accidents. The accident spectrum is designed to encompass a range of possible accident environments, including low-probability accidents that have high consequences and high-probability accidents that have low consequences (i.e., "fender benders"). The collective accident risk results can be directly compared with the incident-free collective risk results because they incorporate the probabilities of accident occurrences.

The RADTRAN 4 calculation of collective accident risk employs models that quantify the range of potential accident severities and the responses of transport packages (i.e., casks) to accident environments. The accident severity spectrum is divided into a number of accident severity categories. Each severity category is assigned a conditional probability of occurrence; that is, the probability that an accident will be of a particular severity if an accident occurs. The more severe the accident, the more remote the chance of such an accident. Release fractions, defined as the fraction of the material in a package that could be released in an accident, are assigned to each accident severity category based on the physical and chemical form of the spent nuclear fuel. The models take into account the transportation mode and the type of packaging being considered. The accident rates, definition of accident severity categories, and release fractions used in this analysis are discussed further in Section E.6.

For accidents involving the release of radioactive material, RADTRAN 4 assumes the material is dispersed in the environment according to standard Gaussian diffusion models. For the risk assessment, default atmospheric dispersion data were used representing an instantaneous ground-level release and a small diameter source cloud (Neuhauser and Kanipe, 1993). The calculation of collective population dose following the release and dispersal of radioactive material includes the following exposure pathways:

- external exposure to the passing radioactive cloud,
- external exposure to contaminated ground,
- internal exposure from inhalation of airborne contaminants, and
- internal exposure from the ingestion of contaminated food.

For the ingestion pathway, state-specific food transfer factors, which relate the amount of radioactive material ingested by people to the amount deposited on the ground, were derived in accordance with the methods described by NRC Guide 1.109 (NRC, 1977b). Radiation doses are calculated using standard dose conversion factors in DOE/EH-0070 (DOE, 1988a) and DOE/EH-0071 (DOE, 1988b).

The collective accident risk for each alternative is determined in a manner similar to that described for incident-free collective risks. Accident risks are first calculated for each unique origin and destination pair ("per-shipment" risk factors) and then summed over all pairs to estimate the total risk for the alternative. The accident risk assessment uses site- and spent nuclear fuel-type-specific radiological and physical characteristics, described further in Section E-6. In addition, the assessment uses route-specific population density information and accident rates derived for individual States.

Accident Consequence Assessment: The RISKIND code is used to provide a detailed assessment of the consequences of the most severe transportation accidents. Whereas the RADTRAN 4 accident risk assessment considers the entire range of accident severities and their related probabilities, the RISKIND accident consequence assessment assumes that an accident of the highest credible severity has occurred. The accident consequence assessment is intended to provide an estimate of the maximum potential impact posed by a severe transportation accident involving spent nuclear fuel.

The severe accidents considered in the consequence assessment are characterized by extreme mechanical and thermal forces. In all cases, these accidents result in a release of radioactive material to the environment. The accidents correspond to those within the highest accident severity category as described above. These accidents represent low-probability, high-consequence events. The probability of accidents of this magnitude occurring for each alternative depends on the total shipment distance. However, accidents of this severity are extremely rare in general.

RISKIND was used for the accident consequence assessment for two reasons. First, the code has the ability to model the complex atmospheric dispersion present in severe accident environments. The atmospheric dispersion is modeled as an instantaneous release using standard Gaussian puff methods. In addition, because severe accidents routinely involve fires, modeling of the potential radiological consequences takes into account physical phenomena resulting from the fire, such as buoyant plume rise. Second, RISKIND can be used to estimate the dose to MEIs in the vicinity of an accident. The location of the MEI is determined by RISKIND based on the atmospheric conditions assumed at the time of the accident and thermal characteristics of the release.

The consequences of the most severe accidents are calculated for both local populations and MEIs. The population dose includes the population within 80 km (50 mi) of the accident site. The exposure pathways considered are similar to those discussed above for the accident risk assessment. Although post-accident remedial activities (e.g., immediate evacuation of the public or cleanup of dispersed radioactive material) would reduce the consequences of an accident, these activities were not given credit in the dose calculations.

Because it is impossible to predict the exact location of a severe transportation accident, separate accident consequences are calculated for accidents occurring in rural, suburban, and urban population density zones. Moreover, to address the effects of the atmospheric conditions existing at the time of an accident, two different atmospheric conditions are considered. The first case assumes neutral atmospheric conditions, and the second, stable conditions. Atmospheric conditions are discussed further in Section E.6.

Nonradiological Accident Risk Assessment: The nonradiological accident risk refers to the potential occurrence of transportation accidents that directly result in fatalities that are not related to the shipment cargo. This risk represents fatalities from mechanical causes. State-specific transportation fatality rates are used in the assessment and are discussed in Section E.6. Nonradiological accident risks are calculated for each alternative by multiplying the total distance traveled in each State by the appropriate State fatality rate. In all cases, the nonradiological accident risks are calculated using round-trip shipment distances.

E.6 Input Parameters and Assumptions

The transportation risk assessment is designed to ensure—through uniform and judicious selection of models, data, and assumptions—that relative comparisons of risk among the various alternatives are meaningful. The major input parameters and assumptions used in the transportation risk assessment are discussed below.

Appendix B lists the casks that are being considered for intersite shipments. The sizes of casks identified vary considerably. Since it is not clear what size of cask would be used for intersite shipments, and since the shipments would not begin until 2005, hypothetical cask sizes are used in this assessment. Additionally, fuel that arrives at an interim site would be physically modified, depending on the dry or wet storage option chosen. Therefore, it is assumed that if spent nuclear fuel were shipped by truck, the number of shipments would be one-quarter of the number of shipments from ports. If the spent nuclear fuel were shipped by rail, the number of shipments would be one-tenth of the number of shipments from ports.

E.6.1 Spent Nuclear Fuel Inventory and Characterization Data

For the purposes of analysis, the foreign research reactor spent nuclear fuel has been characterized into five different spent nuclear fuel categories for shipments into ports and two for shipments between DOE sites. The detailed discussion of the fuel and casks is provided in Appendix B. The curie content of fully loaded shipments is summarized in Table E-5. The approach for calculating the number of shipments from the various countries is shown in Appendix B.

Table E-5 Curie Content of Fully Loaded Shipping Casks for Representative Fuel Types

Isotopes	*Material Type*					
	BR-2	*RHF*	*TRIGA*	*NRU*	*HLW (1 yr)*	*Target*
Tritium	$8.64 \times 10^{+1}$	$3.70 \times 10^{+1}$	$1.31 \times 10^{+1}$	9.48×10^{-1}	~ 0	~ 0
Krypton 85	$2.47 \times 10^{+3}$	$1.07 \times 10^{+3}$	$3.63 \times 10^{+2}$	$2.71 \times 10^{+3}$	~ 0	~ 0
Strontium 89	$4.08 \times 10^{+4}$	$1.76 \times 10^{+4}$	$2.75 \times 10^{+3}$	$9.72 \times 10^{+3}$	$3.07 \times 10^{+6}$	$1.95 \times 10^{+2}$
Strontium 90	$2.08 \times 10^{+4}$	$8.93 \times 10^{+3}$	$3.16 \times 10^{+3}$	$2.32 \times 10^{+4}$	$1.74 \times 10^{+6}$	$1.58 \times 10^{+2}$
Yttrium 90	$2.08 \times 10^{+4}$	$8.93 \times 10^{+3}$	$3.16 \times 10^{+3}$	$2.32 \times 10^{+4}$	$1.74 \times 10^{+6}$	$1.58 \times 10^{+2}$
Yttrium 91	$7.30 \times 10^{+4}$	$3.14 \times 10^{+4}$	$4.56 \times 10^{+3}$	$2.02 \times 10^{+4}$	$5.48 \times 10^{+6}$	$3.69 \times 10^{+2}$
Zirconium 95	$1.07 \times 10^{+5}$	$4.63 \times 10^{+4}$	$6.48 \times 10^{+3}$	$3.38 \times 10^{+4}$	$8.08 \times 10^{+6}$	$5.67 \times 10^{+2}$
Niobium 95	$2.20 \times 10^{+5}$	$9.49 \times 10^{+4}$	$1.28 \times 10^{+4}$	$7.34 \times 10^{+4}$	$1.65 \times 10^{+7}$	$1.21 \times 10^{+3}$
Ruthenium 103	$8.90 \times 10^{+3}$	$3.77 \times 10^{+3}$	$8.44 \times 10^{+2}$	$1.44 \times 10^{+3}$	$7.16 \times 10^{+5}$	$3.57 \times 10^{+1}$
Rhodium 103m	$8.90 \times 10^{+3}$	$3.77 \times 10^{+3}$	$8.44 \times 10^{+2}$	$1.44 \times 10^{+3}$	$7.16 \times 10^{+5}$	$3.57 \times 10^{+1}$
Ruthenium 106	$2.15 \times 10^{+4}$	$9.16 \times 10^{+3}$	$2.54 \times 10^{+3}$	$1.84 \times 10^{+4}$	$1.88 \times 10^{+6}$	$1.49 \times 10^{+2}$
Rhodium 106m	$2.15 \times 10^{+4}$	$9.16 \times 10^{+3}$	$2.54 \times 10^{+3}$	$1.84 \times 10^{+4}$	$1.88 \times 10^{+6}$	$1.49 \times 10^{+2}$
Tin 123	$4.27 \times 10^{+2}$	$1.84 \times 10^{+2}$	$2.71 \times 10^{+1}$	$2.40 \times 10^{+2}$	$5.84 \times 10^{+4}$	$2.70 \times 10^{+0}$
Antimony 125	$8.90 \times 10^{+2}$	$3.81 \times 10^{+2}$	$1.19 \times 10^{+2}$	$9.12 \times 10^{+2}$	$7.57 \times 10^{+4}$	$6.51 \times 10^{+0}$
Tellurium 125m	$2.12 \times 10^{+2}$	$9.06 \times 10^{+1}$	$2.87 \times 10^{+1}$	$2.21 \times 10^{+2}$	$1.81 \times 10^{+4}$	$1.56 \times 10^{+0}$
Tellurium 127m	$8.87 \times 10^{+2}$	$3.82 \times 10^{+2}$	$5.57 \times 10^{+1}$	$4.42 \times 10^{+2}$	$6.97 \times 10^{+4}$	$5.39 \times 10^{+0}$
Tellurium 129m	$1.89 \times 10^{+2}$	$7.98 \times 10^{+1}$	$2.31 \times 10^{+1}$	$2.30 \times 10^{+1}$	$1.59 \times 10^{+4}$	6.73×10^{-1}
Cesium 134	$1.64 \times 10^{+4}$	$4.00 \times 10^{+3}$	$1.16 \times 10^{+3}$	$3.54 \times 10^{+4}$	$1.41 \times 10^{+6}$	6.12×10^{-1}
Cesium-137	$2.06 \times 10^{+4}$	$8.87 \times 10^{+3}$	$3.19 \times 10^{+3}$	$2.30 \times 10^{+4}$	$1.74 \times 10^{+6}$	$1.56 \times 10^{+2}$
Cerium 141	$5.74 \times 10^{+3}$	$2.44 \times 10^{+3}$	$6.97 \times 10^{+2}$	$6.65 \times 10^{+3}$	$5.59 \times 10^{+5}$	$2.03 \times 10^{+1}$
Cerium 144	$3.12 \times 10^{+5}$	$1.35 \times 10^{+5}$	$2.55 \times 10^{+4}$	$2.54 \times 10^{+5}$	$2.49 \times 10^{+7}$	$2.18 \times 10^{+3}$
Praseodymium 144	$3.12 \times 10^{+5}$	$1.35 \times 10^{+5}$	$2.55 \times 10^{+4}$	$2.54 \times 10^{+5}$	$2.49 \times 10^{+7}$	$2.18 \times 10^{+3}$
Promethium 147	$4.83 \times 10^{+4}$	$2.46 \times 10^{+4}$	$7.02 \times 10^{+3}$	$2.98 \times 10^{+4}$	$3.70 \times 10^{+6}$	$5.14 \times 10^{+2}$
Promethium 148m	$7.56 \times 10^{+1}$	$2.92 \times 10^{+1}$	$4.68 \times 10^{+1}$	$1.40 \times 10^{+0}$	$7.13 \times 10^{+3}$	2.43×10^{-2}
Europium 154	$6.20 \times 10^{+2}$	$1.63 \times 10^{+2}$	$4.18 \times 10^{+1}$	$1.35 \times 10^{+3}$	$6.24 \times 10^{+4}$	7.90×10^{-2}
Europium 155	$1.30 \times 10^{+2}$	$4.56 \times 10^{+1}$	$2.27 \times 10^{+1}$	$2.45 \times 10^{+2}$	$1.29 \times 10^{+4}$	$3.35 \times 10^{+0}$
Uranium 234	9.14×10^{-4}	3.74×10^{-4}	1.81×10^{-4}	1.57×10^{-3}	~ 0	6.81×10^{-6}
Uranium 235	1.38×10^{-2}	1.09×10^{-2}	7.91×10^{-3}	6.06×10^{-3}	~ 0	3.98×10^{-3}
Uranium 238	3.41×10^{-4}	2.06×10^{-4}	6.51×10^{-3}	2.67×10^{-5}	~ 0	7.22×10^{-5}
Plutonium 238	$6.42 \times 10^{+1}$	$1.03 \times 10^{+1}$	$3.03 \times 10^{+0}$	$2.70 \times 10^{+2}$	$8.48 \times 10^{+3}$	1.60×10^{-4}
Plutonium 239	$1.84 \times 10^{+0}$	8.89×10^{-2}	5.50×10^{-1}	3.32×10^{-1}	$4.05 \times 10^{+2}$	2.95×10^{-2}
Plutonium 240	$1.20 \times 10^{+0}$	4.21×10^{-1}	$2.09 \times 10^{+0}$	2.42×10^{-1}	$3.26 \times 10^{+2}$	6.85×10^{-4}
Plutonium 241	$2.84 \times 10^{+2}$	$6.77 \times 10^{+1}$	$2.13 \times 10^{+2}$	$7.09 \times 10^{+1}$	$7.84 \times 10^{+4}$	7.09×10^{-3}
Americium 241	3.96×10^{-1}	9.67×10^{-2}	4.07×10^{-1}	1.24×10^{-1}	$9.84 \times 10^{+1}$	1.16×10^{-5}
Americium 242m	1.05×10^{-3}	1.55×10^{-4}	9.00×10^{-3}	6.00×10^{-4}	6.70×10^{-1}	2.13×10^{-10}
Americium 243	4.33×10^{-3}	3.76×10^{-3}	4.38×10^{-4}	3.51×10^{-3}	$1.44 \times 10^{+1}$	1.47×10^{-10}
Curium 244	$1.33 \times 10^{+0}$	9.26×10^{-3}	7.14×10^{-3}	2.70×10^{-1}	$1.22 \times 10^{+2}$	1.63×10^{-10}
Curium 242	$1.75 \times 10^{+0}$	1.27×10^{-1}	$5.25 \times 10^{+0}$	$1.03 \times 10^{+0}$	$9.91 \times 10^{+2}$	6.86×10^{-8}

E.6.2 Shipment External Dose Rates

The dose and corresponding risk to populations and MEIs during incident-free transportation conditions are directly proportional to the assumed shipment external dose rate. The Federal regulations for maximum allowable external dose rates for exclusive-use shipments were presented in Section E.3.

The actual shipment dose rate is a complex function of the composition and configuration of shielding and containment materials used in the cask, the geometry of the loaded shipments, and characteristics of the spent nuclear fuel itself. Based on actual measurements of the dose rate outside real shipping casks, a realistic dose rate of 1 mrem per hr at a distance of 2 m (6.6 ft) was estimated, as described in Appendix F.

However, since individual casks would be expected to exceed this average value, the analysis assumes that all casks would be at the regulatory limits of 10 mrem per hr at 2 m (6.6 ft). In practice, external dose rates would vary from spent nuclear fuel type to spent nuclear fuel type and from shipment to shipment.

E.6.3 Accident Involvement Rates

For the calculation of accident risks, vehicle accident and fatality rates are taken from data provided in other reports (Saricks and Kvitek, 1994). For each transport mode, accident rates are generically defined as the number of accident involvements (or fatalities) in a given year per unit of travel of that mode in that same year. Therefore, the rate is a fractional value, with accident-involvement count as the numerator of the fraction and vehicular activity (total travel distance) as its denominator. Accident rates are generally determined for a multi-year period. For assessment purposes, the total number of expected accidents or fatalities is calculated by multiplying the total shipment distance for a specific case by the appropriate accident or fatality rate.

For truck transportation, the rates presented are specifically for heavy combination trucks involved in interstate commerce (Saricks and Kvitek, 1994). Heavy combination trucks are rigs composed of a separable tractor unit containing the engine and one to three freight trailers connected to each other. Heavy combination trucks are typically used for radioactive waste shipments. The truck accident rates are computed for each State based on statistics compiled by the Department of Transportation Office of Motor Carriers for 1986–1988. Saricks and Kvitek present accident involvement and fatality counts; estimated kilometers of travel by State; and the corresponding average accident involvement, fatality, and injury rates for the 3 years investigated. Fatalities are deaths (including crew members) that are attributable to the accident or that occurred at any time within 30 days thereafter.

Rail accident rates are computed and presented similarly to truck accident rates; however, the unit of haulage is considered to be the railcar (Saricks and Kvitek, 1994). The State-specific rail accident involvement and fatality rates are based on statistics compiled by the Federal Railroad Administration for 1985–1988. Rail accident rates include both main line accidents and those occurring in railyards. It is important to note that the accident rates used in this assessment were computed using the universe of all interstate heavy combination truck shipments, independent of shipment cargo. The cited report points out that shippers and carriers of radioactive material generally have a higher-than-average awareness of transport risk and prepare cargoes and drivers for such shipments accordingly (Saricks and Kvitek, 1994). This preparation should have a twofold effect of reducing component/equipment failure and mitigating the human error contribution to accident causation. These effects were not given credit in the accident assessment.

E.6.4 Cask Accident Response Characteristics

E.6.4.1 Accident Severity Categories

A generic method to characterize the potential severity of transportation accidents was first described in an NRC report commonly referred to as NUREG-0170 (NRC, 1977a). The NRC method divided the spectrum of transportation accident severities into eight categories. Subsequently, other studies have divided the same accident spectrum into 6 categories (Wilmot, 1981) and 20 categories (Fischer et al., 1987). Results from the latter study, which utilizes 20 severity categories and is commonly referred to as the "modal study," are used in this analysis.

The modal study (Fischer et al., 1987) was the result of an initiative taken by the NRC to refine more precisely the analysis presented in NUREG-0170 (NRC, 1977a) for spent nuclear fuel shipping casks. Whereas the NUREG-0170 analysis was primarily performed using best engineering judgments and presumptions concerning cask response, the modal study relies on sophisticated structural and thermal engineering analysis and a probabilistic assessment of the conditions that could be experienced in severe transportation accidents. The modal study results are based on representative spent nuclear fuel casks that were assumed to have been designed, manufactured, operated, and maintained in accordance with national codes and standards. Design parameters of the representative casks were chosen to meet the minimum test criteria specified in 10 CFR Part 71. The study is believed to provide realistic, yet conservative, results for radiological releases under transport accident conditions.

In the modal study, potential accident damage to a cask is categorized according to the magnitude of the mechanical forces (impact) and thermal forces (fire) to which a cask may be subjected during an accident. Because all accidents can be described in these terms, severity is independent of the specific accident sequence. In other words, any sequence of events that results in an accident in which a cask is subjected to forces within a certain range of values is assigned to the accident severity category associated with that range. The accident severity scheme is designed to take into account all potential foreseeable transportation accidents, including accidents with low probability but high consequences and those with high probability but low consequences.

Each severity category actually represents a set of accidents defined by a combination of mechanical and thermal forces. A conditional probability of occurrence—that is, the probability that if an accident occurs, it is of a particular severity—is assigned to each category. The cask response regions and the fractional occurrences by accident severity category are shown in Figure E-14 for truck and rail accidents. Accidents in Region (1,1) are the least severe but most frequent, whereas accidents in Region (4,5) are very severe but very infrequent. To determine the expected frequency of an accident of a given severity, the conditional probability in the category is multiplied by the baseline accident rate. The entire spectrum of accident severities is considered in the accident risk assessment.

As discussed above, the accident consequence assessment only considers the potential impacts from the most severe transportation accidents. In terms of risk, the severity of an accident must be viewed in terms of potential radiological consequences, which are directly proportional to the fraction of the radioactive material within a cask that is released to the environment during the accident. In terms of the modal study accident characterization scheme (Figure E-15), the most severe transportation accidents correspond to those in Regions (4,1), (4,2), (4,3), (4,4), (4,5), (3,5), (2,5), and (1,5). Although these regions span the entire range of mechanical and thermal accident loads considered in the modal study, they are characterized by a single set of release fractions and are therefore considered together in the accident consequence assessment.

The conditional probability of the most severe accidents (i.e., the probability that an accident is of maximum severity, assuming that one has occurred) is found by summing the modal study conditional probabilities for the eight individual accident regions listed above. The resultant overall conditional probability is found to be 0.00000984 for truck transportation and 0.000124 for rail transportation. The stated probabilities encompass the entire spectrum of severe accidents, although over 97 percent of the severe truck accidents and nearly 100 percent of the severe rail accidents actually occur in Region (1,5), which is characterized by high thermal stresses and moderate mechanical stresses.

Structural Response (Maximum Strain on Inner Shell, %)

R (4,1)	R (4,2)	R (4,3)	R (4,4)	R (4,5)
R (3,1)	R (3,2)	R (3,3)	R (3,4)	R (3,5)
R (2,1)	R (2,2)	R (2,3)	R (2,4)	R (2,5)
R (1,1)	R (1,2)	R (1,3)	R (1,4)	R (1,5)

S_3 (30) S_2 (2) S_1 (0.2)

T_1 (500) T_2 (600) T_3 (650) T_4 (1050)

Thermal Response (Lead Mid-Thickness Temperature, °F)

Figure E-14 Matrix of Cask Response Regions for Combined Mechanical and Thermal Loads

Legend:

(P_t) = Probability of occurrence assuming a truck accident occurs.
(P_r) = Probability of occurrence assuming a rail accident occurs.

Structural Response (maximum strain on inner shell, %)

S_3 (30)	R(4,1) (P_t)1.532×10^{-7} (P_r)1.786×10^{-9}	R(4,2) 3.926×10^{-14} 3.290×10^{-13}	R(4,3) 1.495×10^{-14} 2.137×10^{-13}	R(4,4) 7.681×10^{-16} 1.644×10^{-13}	R(4,5) $<1 \times 10^{-16}$ 3.459×10^{-14}
S_2 (2)	R(3,1) (P_t)1.7984×10^{-3} (P_r)5.545×10^{-4}	R(3,2) 1.574×10^{-7} 1.021×10^{-7}	R(3,3) 2.034×10^{-7} 6.634×10^{-8}	R(3,4) 1.076×10^{-7} 5.162×10^{-8}	R(3,5) 4.873×10^{-8} 5.296×10^{-8}
S_1 (0.2)	R(2,1) (P_t)3.8192×10^{-3} (P_r)2.7204×10^{-3}	R(2,2) 2.330×10^{-7} 5.011×10^{-7}	R(2,3) 3.008×10^{-7} 3.255×10^{-7}	R(2,4) 1.592×10^{-7} 2.531×10^{-7}	R(2,5) 7.201×10^{-8} 1.075×10^{-8}
	R(1,1) (P_t)0.994316 (P_r)0.993962	R(1,2) 1.687×10^{-5} 1.2275×10^{-3}	R(1,3) 2.362×10^{-5} 7.9511×10^{-4}	R(1,4) 1.525×10^{-5} 6.140×10^{-4}	R(1,5) 9.570×10^{-6} 1.249×10^{-4}

T_1 (500) T_2 (600) T_3 (650) T_4 (1050)

Thermal Response (lead midthickness temperature, °F)

Figure E-15 Fraction of Truck and Rail Accidents Expected within Each Severity Category, Assuming an Accident Occurs

E.6.4.2 Cask Release Fractions

Radiological consequences are calculated by assigning cask release fractions to each accident severity category. The release fraction is defined as the fraction of the radioactive material in a cask that could be released from the package in a given severity of accident. Release fractions take into account all mechanisms necessary to create a release of radioactive material from a damaged cask to the environment. Release fractions vary according to the spent nuclear fuel type and the physical and chemical characteristics of specific radionuclides within the spent nuclear fuel. For instance, most solid radionuclides are difficult to release in particulate form and are therefore relatively nondispersible. Conversely, gaseous radionuclides are relatively easy to release in the unlikely event that the cask and spent nuclear fuel elements are compromised in an accident.

Cask release fractions are given in Table E-6. Two sets of release fractions were used in the assessment depending on the spent nuclear fuel type, consistent with the SNF&INEL Final EIS (DOE, 1995). Release fractions developed for MTR spent nuclear fuel were used for aluminum-clad fuels including BR-2, RHF, and NRU spent nuclear fuel; Release fractions for TRIGA were used for the PRR-1 spent nuclear fuel.

Table E-6 Release Fractions Spent Nuclear Fuel

Cask Response Region	*Release Fractions*[a]				
	Inert Gas	*Iodine*	*Cesium*	*Ruthenium*	*Particulate*
TRIGA Fuel:					
R(1,1)	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	0.0099	0.000075	0.000006	8.1×10^{-7}	6.0×10^{-8}
R(2,1), R(2,2), R(2,3)	0.03	0.00025	0.00002	0.0000027	2.0×10^{-7}
R(1,4), R(2,4), R(3,4)	0.39	0.0043	0.0002	0.000048	0.000002
R(3,1), R(3,2), R(3,3)	0.33	0.0025	0.0002	0.000027	0.000002
R(1,5), R(2,5), R(3,5), R(4,5), R(4,1), R(4,2), R(4,3), R(4,4)	0.63	0.043	0.002	0.00048	0.00002
Aluminum and Metallic Fuel:[b]					
R(1,1)	0.0	0.0	0.0	0.0	0.0
R(1,2), R(1,3)	0.0099	1.1×10^{-7}	3.0×10^{-8}	4.1×10^{-9}	3.0×10^{-10}
R(2,1), R(2,2), R(2,3)	0.033	3.5×10^{-7}	1.0×10^{-7}	1.4×10^{-8}	1.0×10^{-9}
R(1,4), R(2,4), R(3,4)	0.39	0.000006	0.000001	2.4×10^{-7}	1.0×10^{-8}
R(3,1), R(3,2), R(3,3)	0.33	0.0000035	0.000001	1.4×10^{-7}	1.0×10^{-8}
R(1,5), R(2,5), R(3,5), R(4,5), R(4,1), R(4,2), R(4,3), R(4,4)	0.63	0.00006	0.00001	0.0000024	1.0×10^{-7}

[a] *The fraction of the radioactive material released from a cask to the environment during an accident.*

[b] *These release fractions are applicable to all non-TRIGA, aluminum-clad fuel.*

For waste shipments of material other than spent nuclear fuel, the modal study results are not applicable. Therefore, more conservative release fractions from NUREG-0170 are used for vitrified high-level waste and target material. The NUREG-0170 recommendations for release fractions for Type B casks, regardless of content, are given below:

NUREG-0170 Severity Category	Release Fraction
1	0
2	0
3	0.01
4	0.1
5	1
6	1

Source: NRC, 1977a

The values indicate that in the most severe accidents, 100 percent of the material is released from the cask; a highly conservative assumption for most solid waste forms, and somewhat conservative for a powder or cake-like material. The accident assessment also utilizes the fraction of the release that is aerosolized and the fraction of the aerosol that is respirable. The values for high-level waste and target material (assumed to behave as a loose powdered material) were taken from the recommendations provided in RADTRAN 4. These values are shown below:

Physical Waste Form	Aerosolized Fraction	Respirable Fraction
Vitrified wastes	0.000001	0.05
Chunks (i.e., calcinated target material)	0.01	0.05
Loose powders (i.e., oxidized target material)	0.1	0.05

Source: Neuhauser and Kanipe, 1993

Therefore, the maximum total respirable release fraction for the most severe accidents is $5x10^{-8}$ for high-level waste shipments and 0.005 for shipments of target material. The values shown above have been used in the accident calculations for shipments of target material and vitrified material for the foreign research reactor spent nuclear fuel EIS.

E.6.5 Atmospheric Conditions

Radioactive material released to the atmosphere is transported by the wind. The amount of dispersion, or dilution, of the radioactive material concentrations in the air depends on the meteorological conditions at the time of the accident. Because it is impossible to predict the specific location of an overland transportation accident, generic atmospheric conditions were selected for the accident risk and consequence assessments.

For the accident risk assessment, neutral weather conditions (Pasquill Stability Class D with a wind speed of 4 m per sec or 9 mph) were assumed. Since neutral meteorological conditions are the most frequently occurring atmospheric stability condition in the United States, they are most likely to be present in the event of an accident involving a spent nuclear fuel shipment. On the basis of observations from National Weather Service surface meteorological stations at over 300 locations in the United States, on an annual average, neutral conditions (Pasquill Classes C and D) occur about half (50 percent) of the time, while stable (Pasquill Classes E and F) and unstable (Pasquill Class A and B) conditions occur about one-third (33 percent) and one-sixth (17 percent) of the time, respectively (Doty et al., 1976). The neutral category predominates in all seasons, but most frequently in the winter (nearly 60 percent of the observations).

For the accident consequence assessment, doses were assessed under both neutral (Pasquill Stability Class D with a wind speed of 4 m per sec or 9 mph) and stable (Pasquill Stability Class F with a wind speed of 1 m per sec or 2.4 mph) atmospheric conditions. The results calculated for neutral conditions represent the most likely consequences, and the results for stable conditions represent a "worst-case" weather situation.

E.6.6 Health Risk Conversion Factors

The health risk conversion factors used to estimate expected cancer fatalities were taken from International Commission on Radiation Protection Publication 60 (ICRP, 1991): 0.0005 and 0.0004 fatal cancer cases per person-rem for members of the public and workers, respectively. Cancer fatalities and incidence occur over the lifetimes of the exposed populations, and thus are called LCF.

E.6.7 Maximally Exposed Individual Exposure Scenarios

The risk to MEIs has been estimated for a number of hypothetical exposure scenarios during overland transportation using the RISKIND code. The receptors include crew members, departure inspectors, and members of the public exposed during traffic obstructions (traffic jam), while working at a service station, or by living near a port of entry or management site. The dose and risk to MEIs were calculated for given distances and durations of exposure. The distances and durations of exposure for each receptor are similar to those given in previous transportation assessments (DOE, 1987b; DOE, 1995), and are believed to be realistic but conservative. The exposure scenarios considered are the following:

- Crew Members: Truck and rail crew members are not assumed to be occupational radiation workers. Dose estimates are based on realistic locations and estimated travel time, and no credit is taken for shielding in addition to the cask.

- Inspectors (truck and rail): Inspectors are assumed to be either Federal or State vehicle inspectors, and are not assumed to be monitored by a dosimetry program. An average exposure distance of 3 m (10 ft) and an exposure time of 30 minutes (min) is assumed.

- Rail Yard Crew Member: A rail yard crew member is not assumed to be monitored by a dosimetry program. An average exposure distance of 10 m (33 ft) and an exposure time of 2 hr is assumed.

- Resident (truck and rail): A resident is assumed to live 30 m (100 ft) from a port or management site entrance route (truck or rail). Shipments are assumed to pass at a velocity of 24 km per hr (15 mph), and the resident is assumed to be exposed unshielded (i.e., no shielding in addition to the cask, such as that afforded by a structure.) Cumulative doses are assessed for each alternative based on the number of shipments entering or exiting the site and assuming the resident is present for 100 percent of the shipments.

- Person in Traffic Obstruction (truck and rail): A person is assumed to be stopped next to a spent nuclear fuel shipment (due to traffic, etc.). The person is assumed to be exposed (no credit is taken for radiation blocked by the individual's vehicle) at a distance of 1 m (3.3 ft) for a duration of 30 min.

- Person at a Truck Service Station: A person is assumed to be exposed at an average distance of 20 m (66 ft) for a duration of 2 hr. This receptor could be a worker at a truck stop, or a member of the public stopped at the same location.

- Resident Near a Rail Stop: A resident is assumed to live near a rail classification yard. The resident is assumed to be exposed unshielded at a distance of 200 m (660 ft) for a duration of 20 hr.

The largest uncertainty in predicting the dose to MEIs during transportation involves determining the frequency of exposure occurrences. This difficulty results from the uncertainties in future shipment schedules, route selection, and the inherent uncertainty in predicting the frequency of random or chance events. For instance, it is conceivable that an individual could be stopped in traffic next to a shipment of foreign research reactor spent nuclear fuel; however, it is difficult to predict how often the same individual would experience this event. Therefore, for the majority of receptors considered, doses are assessed on a per-event basis. To account for possible multiple exposures, ranges of realistic total doses are discussed qualitatively. One exception is the calculation of the dose to a hypothetical resident living near a port of entry or management site entrance route. For these residents, total doses are calculated based on the number of shipments entering or exiting each site for each of the alternatives.

E.6.8 General RADTRAN Input Parameters

In addition to the specific parameters discussed above, values for a number of general parameters must be specified within the RADTRAN code. These general parameters define basic shipment and traffic characteristics and are specific to the mode of transportation. The RADTRAN code user's manual (Neuhauser and Kanipe, 1993) contains derivations and descriptions of these parameters. The general RADTRAN input parameters used in the transportation risk assessment are summarized in Table E-7.

Table E-7 Summary of General RADTRAN Input Parameters

Parameter	Truck	Rail
Package type	Type B Cask	Type B Cask
Package dimension	3.2 m (10.6 ft)	3.2 m (10.6 ft)
Number of crewmen	2	5
Distance from source to crew	3 m (9.9 ft)	152 m (501.6 ft)
Velocity		
Rural	88 km/hr (55 mph)	64 km/hr (40 mph)
Suburban	40 km/hr (25 mph)	40 km/hr (25 mph)
Urban	24 km/hr (15 mph)	24 km/hr (15 mph)
Stop time per kilometer	0.011 hr/km (0.018 hr/mi)	0.033 hr/km (0.053 hr/mi)
Number of people exposed while stopped	50	Based on Suburban Population Density
Number of people per vehicle sharing route	2	3
Population densities (persons/km^2)	Route Specific (see Table E-4)	Route Specific (see Table E-4)
One-way traffic count (vehicles/hr)		
Rural	470	1
Suburban	780	5
Urban	2,800	5
Cask inventory (Ci)	(see Table E-5)	(see Table E-5)
Accident release fractions	(see Table E-6)	(see Table E-6)
Accident conditional probabilities	(see Figure E-15)	(see Figure E-15)

Source: Neuhauser and Kanipe, 1993.

E.7 Risk Assessment Results

In this section, the risk assessment results are presented for the ports of entry and management sites being considered. The collective population risk results are presented in Section E.7.1. First, the per-shipment risks results are presented in Section E.7.1.1. Then, in Section E.7.1.2, the results are analyzed, evaluated, and simplified so the different program alternatives and options can be evaluated in Section E.7.2.

The risks to MEIs during incident-free transportation conditions are provided in Section E.7.3. The accident consequence results calculated for the most severe transportation accidents are presented in Section E.7.4 for both collective populations and MEIs.

E.7.1 Collective Population Risk Results

E.7.1.1 Per-Shipment Risk Factors

Per-shipment risk factors have been calculated for the collective populations of exposed persons for shipments between all representative ports of entry and the five management sites. Results were calculated for both truck and rail modes, assuming that one cask would be shipped per truck or rail car. Additionally, the risk factors for the ports of Elizabeth, NJ; Philadelphia, PA; and Long Beach, CA are included to show the effect of using high population ports. Risk factors are included for some site-to-site routes, even though there are no shipments anticipated on these routes.

The radiological risks are presented in terms of dose per shipment for each unique route. The doses can be converted to health risks using the International Commission on Radiological Protection Publication 60 conversion factors described in Section E.6.6 (ICRP, 1991). The radiological dose per shipment factors for incident-free transportation conditions are presented in Table E-8 for crew members and members of the general public. The tabulated incident-free doses are based on the external dose rate which is conservatively assumed to be at the regulatory limit of 10 mrem per hr at 2 m. The radiological dose risk factors for accident transportation conditions are presented in Table E-9. The accident risk factors are referred to as "dose risk" because the values incorporate the spectrum of accident severity probabilities and the associated release fractions.

Table E-8 Incident-Free Dose per Shipment for All Spent Nuclear Fuel Types (Person-Rem/Shipment)[a]

Shipments to Hanford Site:						
Route(s)		***Crew***	***General Public***			
			Off-Link	***On-Link***	***Stops***	***Total***
From Eastern Ports						
Charleston, SC (NWS)	Truck	2.50×10^{-1}	9.26×10^{-3}	3.96×10^{-2}	5.92×10^{-1}	6.41×10^{-1}
	Rail	6.33×10^{-2}	2.91×10^{-2}	1.13×10^{-3}	1.70×10^{-2}	4.73×10^{-2}
Charleston, SC (Wando Terminal)	Truck	2.51×10^{-1}	9.61×10^{-3}	4.03×10^{-2}	5.95×10^{-1}	6.45×10^{-1}
	Rail	6.33×10^{-2}	2.91×10^{-2}	1.13×10^{-3}	1.70×10^{-2}	4.73×10^{-2}
Elizabeth, NJ	Truck	2.49×10^{-1}	9.96×10^{-3}	4.10×10^{-2}	5.82×10^{-1}	6.33×10^{-1}
	Rail	6.24×10^{-2}	6.03×10^{-2}	1.66×10^{-3}	1.68×10^{-2}	7.88×10^{-2}
Galveston, TX	Truck	2.05×10^{-1}	1.00×10^{-2}	3.77×10^{-2}	4.82×10^{-1}	5.30×10^{-1}
	Rail	5.20×10^{-2}	1.31×10^{-2}	6.59×10^{-4}	1.51×10^{-2}	2.88×10^{-2}
Jacksonville, FL	Truck	2.60×10^{-1}	1.09×10^{-2}	4.35×10^{-2}	6.05×10^{-1}	6.60×10^{-1}
	Rail	6.35×10^{-2}	2.58×10^{-2}	1.08×10^{-3}	1.65×10^{-2}	4.34×10^{-2}
Newport News, VA	Truck	2.57×10^{-1}	1.16×10^{-2}	4.44×10^{-2}	6.02×10^{-1}	6.58×10^{-1}
	Rail	6.38×10^{-2}	4.02×10^{-2}	1.25×10^{-3}	1.59×10^{-2}	5.73×10^{-2}

Shipments to Hanford Site:							
Route(s)		Crew	General Public				
			Off-Link	On-Link	Stops	Total
Norfolk, VA	Truck	$2.62x10^{-1}$	$1.19x10^{-2}$	$4.50x10^{-2}$	$6.10x10^{-1}$	$6.67x10^{-1}$
	Rail	$6.54x10^{-2}$	$4.10x10^{-2}$	$1.28x10^{-3}$	$1.67x10^{-2}$	$5.90x10^{-2}$
Philadelphia, PA	Truck	$2.58x10^{-1}$	$1.06x10^{-2}$	$4.31x10^{-2}$	$5.94x10^{-1}$	$6.47x10^{-1}$
	Rail	$6.16x10^{-2}$	$5.90x10^{-2}$	$1.58x10^{-3}$	$1.68x10^{-2}$	$7.74x10^{-2}$
Portsmouth, VA	Truck	$2.61x10^{-1}$	$1.24x10^{-2}$	$4.60x10^{-2}$	$6.07x10^{-1}$	$6.65x10^{-1}$
	Rail	$6.49x10^{-2}$	$3.99x10^{-2}$	$1.25x10^{-3}$	$1.64x10^{-2}$	$5.76x10^{-2}$
Savannah, GA	Truck	$2.48x10^{-1}$	$1.03x10^{-2}$	$4.13x10^{-2}$	$5.82x10^{-1}$	$6.34x10^{-1}$
	Rail	$6.38x10^{-2}$	$2.48x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.24x10^{-2}$
MOTSU, NC	Truck	$2.50x10^{-1}$	$9.26x10^{-3}$	$4.00x10^{-2}$	$5.94x10^{-1}$	$6.43x10^{-1}$
	Rail	$6.57x10^{-2}$	$2.85x10^{-2}$	$1.20x10^{-3}$	$1.71x10^{-2}$	$4.68x10^{-2}$
Wilmington, NC	Truck	$2.59x10^{-1}$	$9.34x10^{-3}$	$4.07x10^{-2}$	$6.13x10^{-1}$	$6.63x10^{-1}$
	Rail	$6.56x10^{-2}$	$2.84x10^{-2}$	$1.19x10^{-3}$	$1.71x10^{-2}$	$4.67x10^{-2}$
From Western Ports						
NWS Concord, CA	Truck	$8.06x10^{-2}$	$4.55x10^{-3}$	$1.55x10^{-2}$	$1.77x10^{-1}$	$1.97x10^{-1}$
	Rail	$2.77x10^{-2}$	$1.88x10^{-2}$	$4.83x10^{-4}$	$8.77x10^{-3}$	$2.81x10^{-2}$
Long Beach, CA	Truck	$1.19x10^{-1}$	$1.13x10^{-2}$	$2.98x10^{-2}$	$2.57x10^{-1}$	$2.98x10^{-1}$
	Rail	$3.84x10^{-2}$	$3.71x10^{-2}$	$7.16x10^{-4}$	$1.46x10^{-2}$	$5.25x10^{-2}$
Portland, OR	Truck	$2.35x10^{-2}$	$1.50x10^{-3}$	$4.90x10^{-3}$	$5.24x10^{-2}$	$5.88x10^{-2}$
	Rail	$1.57x10^{-2}$	$4.36x10^{-3}$	$1.12x10^{-4}$	$7.04x10^{-3}$	$1.15x10^{-2}$
Tacoma, WA	Truck	$2.50x10^{-2}$	$1.73x10^{-3}$	$5.45x10^{-3}$	$5.14x10^{-2}$	$5.85x10^{-2}$
	Rail	$1.80x10^{-2}$	$5.69x10^{-3}$	$1.82x10^{-4}$	$6.12x10^{-3}$	$1.20x10^{-2}$
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	$2.49x10^{-1}$	$1.05x10^{-2}$	$4.23x10^{-2}$	$5.73x10^{-1}$	$6.26x10^{-1}$
	Rail	$6.02x10^{-2}$	$5.10x10^{-2}$	$1.40x10^{-3}$	$1.65x10^{-2}$	$6.89x10^{-2}$
Idaho National Engineering Laboratory	Truck	$4.92x10^{-1}$	$1.42x10^{-3}$	$7.63x10^{-3}$	$1.24x10^{-1}$	$1.33x10^{-1}$
	Rail	$2.28x10^{-2}$	$3.88x10^{-3}$	$1.76x10^{-4}$	$7.64x10^{-3}$	$1.17x10^{-2}$
Nevada Test Site	Truck	$9.91x10^{-2}$	$5.37x10^{-3}$	$1.92x10^{-2}$	$2.34x10^{-1}$	$2.58x10^{-1}$
	Rail	$3.36x10^{-1}$	$6.24x10^{-3}$	$3.08x10^{-4}$	$1.12x10^{-2}$	$1.77x10^{-2}$
Oak Ridge Reservation	Truck	$2.10x10^{-1}$	$7.29x10^{-3}$	$3.31x10^{-2}$	$5.10x10^{-1}$	$5.50x10^{-1}$
	Rail	$5.56x10^{-2}$	$1.64x10^{-2}$	$6.92x10^{-4}$	$1.60x10^{-1}$	$3.31x10^{-2}$
Savannah River	Truck	$2.42x10^{-1}$	$1.02x10^{-2}$	$4.01x10^{-2}$	$5.65x10^{-1}$	$6.15x10^{-1}$
	Rail	$6.15x10^{-2}$	$2.74x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.51x10^{-2}$
Sweetgrass, MT	Truck	$7.27x10^{-1}$	$1.76x10^{-3}$	$1.03x10^{-2}$	$1.81x10^{-1}$	$1.93x10^{-1}$
	Rail	$2.19x10^{-2}$	$3.81x10^{-3}$	$1.62x10^{-4}$	$7.83x10^{-3}$	$1.18x10^{-2}$

Shipments to Idaho National Engineering Laboratory:						
Route(s)		Crew	General Public			
			Off-Link	On-Link	Stops	Total
From Eastern Ports						
Charleston, SC (NWS)	Truck	$2.16x10^{-1}$	$8.41x10^{-3}$	$3.48x10^{-2}$	$5.05x10^{-1}$	$5.49x10^{-1}$
	Rail	$5.41x10^{-2}$	$2.68x10^{-2}$	$1.01x10^{-3}$	$1.45x10^{-2}$	$4.24x10^{-2}$
Charleston, SC (Wando Terminal)	Truck	$2.18x10^{-1}$	$8.76x10^{-3}$	$3.56x10^{-2}$	$5.09x10^{-1}$	$5.53x10^{-1}$
	Rail	$5.41x10^{-2}$	$2.68x10^{-2}$	$1.01x10^{-3}$	$1.45x10^{-2}$	$4.24x10^{-2}$
Elizabeth, NJ	Truck	$2.15x10^{-1}$	$9.11x10^{-3}$	$3.59x10^{-2}$	$4.96x10^{-1}$	$5.41x10^{-1}$
	Rail	$5.32x10^{-2}$	$5.81x10^{-2}$	$1.53x10^{-3}$	$1.44x10^{-2}$	$7.40x10^{-2}$
Galveston, TX	Truck	$1.71x10^{-1}$	$9.18x10^{-3}$	$3.26x10^{-2}$	$3.96x10^{-1}$	$4.37x10^{-1}$
	Rail	$4.28x10^{-2}$	$1.08x10^{-2}$	$5.35x10^{-4}$	$1.24x10^{-2}$	$2.38x10^{-2}$
Jacksonville, FL	Truck	$2.26x10^{-1}$	$9.78x10^{-3}$	$3.77x10^{-2}$	$5.18x10^{-1}$	$5.66x10^{-1}$
	Rail	$5.42x10^{-2}$	$2.35x10^{-2}$	$9.54x10^{-4}$	$1.40x10^{-2}$	$3.85x10^{-2}$
Newport News, VA	Truck	$2.24x10^{-1}$	$1.08x10^{-2}$	$3.93x10^{-2}$	$5.16x10^{-1}$	$5.66x10^{-1}$
	Rail	$5.45x10^{-2}$	$3.79x10^{-2}$	$1.13x10^{-3}$	$1.35x10^{-2}$	$5.25x10^{-2}$

Shipments to Idaho National Engineering Laboratory:							
Route(s)		***Crew***	***General Public***				
			Off-Link	***On-Link***	***Stops***	***Total***	
From Eastern Ports							
Norfolk, VA	Truck	2.28×10^{-1}	1.08×10^{-2}	3.93×10^{-2}	5.24×10^{-1}	5.74×10^{-1}	
	Rail	5.62×10^{-2}	3.87×10^{-2}	1.16×10^{-3}	1.43×10^{-2}	5.42×10^{-2}	
Philadelphia, PA	Truck	2.24×10^{-1}	9.71×10^{-3}	3.80×10^{-2}	5.08×10^{-1}	5.55×10^{-1}	
	Rail	5.24×10^{-2}	5.67×10^{-2}	1.46×10^{-3}	1.44×10^{-2}	7.25×10^{-2}	
Portsmouth, VA	Truck	2.27×10^{-1}	1.15×10^{-2}	4.09×10^{-2}	5.20×10^{-1}	5.73×10^{-1}	
	Rail	5.57×10^{-2}	3.76×10^{-2}	1.13×10^{-3}	1.40×10^{-2}	5.27×10^{-2}	
Savannah, GA	Truck	2.15×10^{-1}	9.41×10^{-3}	3.62×10^{-2}	4.96×10^{-1}	5.42×10^{-1}	
	Rail	5.46×10^{-2}	2.25×10^{-2}	9.53×10^{-4}	1.41×10^{-2}	3.75×10^{-2}	
MOTSU, NC	Truck	2.11×10^{-1}	7.80×10^{-3}	3.35×10^{-2}	4.98×10^{-1}	5.40×10^{-1}	
	Rail	5.65×10^{-2}	2.62×10^{-2}	1.07×10^{-3}	1.47×10^{-2}	4.20×10^{-2}	
Wilmington, NC	Truck	2.25×10^{-1}	8.50×10^{-3}	3.56×10^{-2}	5.27×10^{-1}	5.71×10^{-1}	
	Rail	5.63×10^{-2}	2.61×10^{-2}	1.07×10^{-3}	1.47×10^{-2}	4.19×10^{-2}	
From Western Ports							
NWS Concord, CA	Truck	8.40×10^{-2}	4.84×10^{-3}	1.72×10^{-2}	1.95×10^{-1}	2.17×10^{-1}	
	Rail	2.71×10^{-2}	8.71×10^{-3}	2.96×10^{-4}	7.88×10^{-3}	1.69×10^{-2}	
Long Beach, CA	Truck	9.93×10^{-2}	1.28×10^{-2}	3.06×10^{-2}	2.03×10^{-1}	2.46×10^{-1}	
	Rail	2.92×10^{-2}	3.48×10^{-2}	5.92×10^{-4}	1.20×10^{-2}	4.74×10^{-2}	
Portland, OR	Truck	6.27×10^{-2}	2.46×10^{-3}	1.06×10^{-2}	1.53×10^{-1}	1.66×10^{-1}	
	Rail	2.49×10^{-2}	5.01×10^{-3}	2.01×10^{-4}	7.23×10^{-3}	1.24×10^{-2}	
Tacoma, WA	Truck	7.04×10^{-2}	2.71×10^{-3}	1.19×10^{-2}	1.69×10^{-1}	1.83×10^{-1}	
	Rail	2.74×10^{-2}	8.57×10^{-3}	3.04×10^{-4}	7.71×10^{-3}	1.66×10^{-2}	
From DOE Sites/Canadian Border							
Alexandria Bay, NY	Truck	2.16×10^{-1}	9.64×10^{-3}	3.72×10^{-2}	4.87×10^{-1}	5.34×10^{-1}	
	Rail	5.10×10^{-2}	4.87×10^{-2}	1.28×10^{-3}	1.41×10^{-2}	6.40×10^{-2}	
Hanford Site	Truck	4.92×10^{-1}	1.42×10^{-2}	7.63×10^{-2}	1.24×10^{-1}	1.33×10^{-1}	
	Rail	2.28×10^{-1}	3.88×10^{-2}	1.76×10^{-3}	7.64×10^{-2}	1.17×10^{-2}	
Nevada Test Site	Truck	6.56×10^{-1}	4.52×10^{-2}	1.40×10^{-1}	1.47×10^{-1}	1.66×10^{-1}	
	Rail	2.44×10^{-1}	3.95×10^{-2}	1.84×10^{-3}	8.29×10^{-2}	1.24×10^{-2}	
Oak Ridge Reservation	Truck	1.76×10^{-1}	6.45×10^{-1}	2.80×10^{-1}	4.24×10^{-1}	4.58×10^{-1}	
	Rail	4.63×10^{-1}	1.41×10^{-1}	5.69×10^{-3}	1.33×10^{-1}	2.80×10^{-2}	
Savannah River	Truck	2.08×10^{-1}	9.34×10^{-2}	3.50×10^{-1}	4.79×10^{-1}	5.23×10^{-1}	
	Rail	5.23×10^{-1}	2.51×10^{-1}	9.56×10^{-3}	1.41×10^{-1}	4.02×10^{-2}	
Sweetgrass, MT	Truck	4.25×10^{-1}	6.45×10^{-3}	5.67×10^{-2}	1.12×10^{-1}	1.19×10^{-1}	
	Rail	3.24×10^{-1}	5.37×10^{-2}	2.84×10^{-3}	9.14×10^{-2}	1.48×10^{-2}	

Shipments to Nevada Test Site:						
Route(s)		***Crew***	***General Public***			
			Off-Link	***On-Link***	***Stops***	***Total***
From Eastern Ports						
Charleston, SC (NWS)	Truck	2.25×10^{-1}	9.15×10^{-3}	3.69×10^{-2}	5.27×10^{-1}	5.73×10^{-1}
	Rail	6.13×10^{-2}	2.95×10^{-2}	1.11×10^{-3}	1.69×10^{-2}	4.75×10^{-2}
Charleston, SC (Wando Terminal)	Truck	2.27×10^{-1}	9.50×10^{-3}	3.76×10^{-2}	5.30×10^{-1}	5.77×10^{-1}
	Rail	6.13×10^{-2}	2.95×10^{-2}	1.11×10^{-3}	1.69×10^{-2}	4.75×10^{-2}
Elizabeth, NJ	Truck	2.48×10^{-1}	1.35×10^{-2}	4.60×10^{-2}	5.53×10^{-1}	6.13×10^{-1}
	Rail	6.05×10^{-2}	6.07×10^{-3}	1.63×10^{-3}	1.66×10^{-2}	7.8×10^{-2}
Jacksonville, FL	Truck	2.35×10^{-1}	1.10×10^{-2}	4.07×10^{-2}	5.40×10^{-1}	5.91×10^{-1}
	Rail	6.15×10^{-2}	2.61×10^{-2}	1.05×10^{-3}	1.64×10^{-2}	4.36×10^{-2}
Newport News, VA	Truck	2.33×10^{-1}	1.20×10^{-2}	4.23×10^{-2}	5.37×10^{-1}	5.92×10^{-1}
	Rail	6.18×10^{-2}	4.05×10^{-2}	1.22×10^{-3}	1.57×10^{-2}	5.75×10^{-2}

Shipments to Nevada Test Site:							
Route(s)		***Crew***	***General Public***			
			Off-Link	***On-Link***	***Stops***	***Total***
Norfolk, VA	Truck	2.73×10^{-1}	1.19×10^{-2}	4.23×10^{-2}	5.45×10^{-1}	5.99×10^{-1}
	Rail	6.35×10^{-2}	4.14×10^{-2}	1.2×10^{-3}	1.66×10^{-2}	5.92×10^{-2}
Philadelphia, PA	Truck	2.44×10^{-1}	1.30×10^{-2}	4.49×10^{-2}	5.43×10^{-1}	6.01×10^{-1}
	Rail	5.97×10^{-2}	5.94×10^{-2}	1.56×10^{-3}	1.66×10^{-2}	7.75×10^{-2}
Portsmouth, VA	Truck	2.36×10^{-1}	1.27×10^{-2}	4.39×10^{-2}	5.42×10^{-1}	5.98×10^{-1}
	Rail	6.30×10^{-2}	4.3×10^{-2}	1.23×10^{-3}	1.63×10^{-2}	5.78×10^{-2}
Savannah, GA	Truck	2.24×10^{-1}	1.06×10^{-2}	3.92×10^{-2}	5.18×10^{-1}	5.67×10^{-1}
	Rail	6.19×10^{-2}	2.51×10^{-2}	1.05×10^{-3}	1.64×10^{-2}	4.26×10^{-2}
MOTSU, NC	Truck	2.22×10^{-1}	1.06×10^{-2}	3.95×10^{-2}	5.09×10^{-1}	5.59×10^{-1}
	Rail	6.38×10^{-2}	2.89×10^{-2}	1.17×10^{-3}	1.70×10^{-2}	4.70×10^{-2}
Wilmington, NC	Truck	2.35×10^{-1}	9.67×10^{-3}	3.86×10^{-2}	5.49×10^{-1}	5.97×10^{-1}
	Rail	6.36×10^{-2}	2.88×10^{-2}	1.16×10^{-3}	1.70×10^{-2}	4.69×10^{-2}
From Western Ports						
NWS Concord, CA	Truck	6.88×10^{-2}	8.04×10^{-3}	1.99×10^{-2}	1.47×10^{-1}	1.75×10^{-1}
	Rail	2.60×10^{-2}	1.83×10^{-2}	4.73×10^{-4}	8.17×10^{-3}	2.70×10^{-2}
Long Beach, CA	Truck	4.63×10^{-2}	9.79×10^{-3}	2.04×10^{-2}	8.30×10^{-2}	1.13×10^{-1}
	Rail	1.98×10^{-2}	3.09×10^{-2}	4.37×10^{-4}	9.63×10^{-3}	4.10×10^{-2}
Portland, OR	Truck	1.13×10^{-1}	6.62×10^{-3}	2.28×10^{-2}	2.63×10^{-1}	2.92×10^{-1}
	Rail	3.58×10^{-2}	7.38×10^{-3}	3.33×10^{-1}	1.10×10^{-2}	1.87×10^{-2}
Tacoma, WA	Truck	1.20×10^{-1}	6.66×10^{-3}	2.34×10^{-2}	2.78×10^{-1}	3.08×10^{-1}
	Rail	3.83×10^{-2}	1.09×10^{-2}	4.36×10^{-4}	1.11×10^{-2}	2.24×10^{-2}
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	2.39×10^{-1}	1.11×10^{-2}	4.21×10^{-2}	5.42×10^{-1}	5.95×10^{-1}
	Rail	5.83×10^{-2}	5.13×10^{-2}	1.37×10^{-3}	1.63×10^{-2}	6.90×10^{-2}
Hanford Site	Truck	9.91×10^{-2}	5.37×10^{-3}	1.92×10^{-2}	2.34×10^{-1}	2.58×10^{-1}
	Rail	3.36×10^{-2}	6.24×10^{-3}	3.08×10^{-1}	1.12×10^{-2}	1.77×10^{-2}
Idaho National Engineering Laboratory	Truck	6.56×10^{-2}	4.52×10^{-3}	1.40×10^{-2}	1.47×10^{-1}	1.66×10^{-1}
	Rail	2.44×10^{-2}	3.95×10^{-3}	1.84×10^{-1}	8.29×10^{-3}	1.24×10^{-2}
Oak Ridge Reservation	Truck	1.86×10^{-1}	7.6×10^{-3}	3.10×10^{-2}	4.45×10^{-1}	4.84×10^{-1}
	Rail	5.36×10^{-2}	1.68×10^{-2}	6.66×10^{-1}	1.63×10^{-2}	3.37×10^{-2}
Savannah River	Truck	2.17×10^{-1}	1.05×10^{-2}	3.80×10^{-2}	5.00×10^{-1}	5.48×10^{-1}
	Rail	5.96×10^{-2}	2.77×10^{-2}	1.05×10^{-3}	1.65×10^{-2}	4.53×10^{-2}

Shipments to Oak Ridge Reservation:						
Route(s)		***General Public***				
		Crew	***Off-Link***	***On-Link***	***Stops***	***Total***
From Eastern Ports						
Charleston, SC (NWS)	Truck	3.98×10^{-2}	1.73×10^{-3}	6.11×10^{-3}	8.32×10^{-2}	9.11×10^{-2}
	Rail	2.15×10^{-2}	6.88×10^{-3}	3.60×10^{-4}	5.33×10^{-3}	1.26×10^{-2}
Charleston, SC (Wando Terminal)	Truck	4.18×10^{-2}	2.09×10^{-3}	6.85×10^{-3}	8.63×10^{-2}	9.53×10^{-2}
	Rail	2.15×10^{-2}	6.88×10^{-3}	3.60×10^{-4}	5.33×10^{-3}	1.26×10^{-2}
Elizabeth, NJ	Truck	8.02×10^{-2}	4.85×10^{-3}	1.45×10^{-2}	1.53×10^{-1}	1.72×10^{-1}
	Rail	2.49×10^{-2}	4.30×10^{-2}	9.20×10^{-4}	7.73×10^{-3}	5.17×10^{-2}
Galveston, TX	Truck	9.53×10^{-2}	5.71×10^{-3}	1.73×10^{-2}	1.99×10^{-1}	2.22×10^{-1}
	Rail	2.94×10^{-2}	1.87×10^{-2}	6.28×10^{-4}	8.98×10^{-3}	2.83×10^{-2}
Jacksonville, FL	Truck	5.88×10^{-2}	2.88×10^{-3}	9.64×10^{-3}	1.17×10^{-1}	1.30×10^{-1}
	Rail	2.12×10^{-2}	8.80×10^{-3}	3.59×10^{-4}	6.51×10^{-3}	1.57×10^{-2}
Newport News, VA	Truck	5.68×10^{-2}	3.57×10^{-3}	1.09×10^{-2}	1.15×10^{-1}	1.29×10^{-1}
	Rail	2.45×10^{-2}	1.11×10^{-2}	5.45×10^{-4}	6.00×10^{-3}	1.77×10^{-2}
Norfolk, VA	Truck	5.63×10^{-2}	2.85×10^{-3}	9.27×10^{-3}	1.14×10^{-1}	1.26×10^{-1}
	Rail	2.33×10^{-2}	8.51×10^{-3}	4.56×10^{-4}	5.63×10^{-3}	1.46×10^{-2}

Shipments to Oak Ridge Reservation:						
Route(s)		***General Public***				
		Crew	***Off-Link***	***On-Link***	***Stops***	***Total***
Philadelphia, PA	Truck	7.36×10^{-2}	5.74×10^{-3}	1.56×10^{-2}	1.41×10^{-1}	1.62×10^{-1}
	Rail	2.35×10^{-2}	2.83×10^{-2}	7.16×10^{-4}	6.98×10^{-3}	3.60×10^{-2}
Portsmouth, VA	Truck	6.01×10^{-2}	4.32×10^{-3}	1.25×10^{-2}	1.19×10^{-1}	1.36×10^{-1}
	Rail	2.28×10^{-2}	7.44×10^{-3}	4.30×10^{-4}	5.31×10^{-3}	1.32×10^{-2}
Savannah, GA	Truck	4.33×10^{-2}	1.61×10^{-3}	6.44×10^{-3}	9.29×10^{-2}	1.01×10^{-1}
	Rail	2.16×10^{-2}	7.78×10^{-3}	3.58×10^{-4}	6.58×10^{-3}	1.47×10^{-2}
MOTSU, NC	Truck	4.88×10^{-2}	2.07×10^{-3}	7.52×10^{-3}	1.03×10^{-1}	1.12×10^{-1}
	Rail	2.08×10^{-2}	6.52×10^{-3}	3.63×10^{-4}	5.04×10^{-3}	1.19×10^{-2}
Wilmington, NC	Truck	5.00×10^{-2}	2.06×10^{-3}	7.70×10^{-3}	1.05×10^{-1}	1.15×10^{-1}
	Rail	2.06×10^{-2}	6.41×10^{-3}	3.55×10^{-4}	5.00×10^{-3}	1.18×10^{-2}
From Western Ports						
NWS Concord, CA	Truck	2.26×10^{-1}	1.32×10^{-2}	4.47×10^{-2}	5.29×10^{-1}	5.87×10^{-1}
	Rail	5.91×10^{-2}	2.76×10^{-2}	9.50×10^{-4}	1.71×10^{-2}	4.57×10^{-2}
Long Beach, CA	Truck	2.03×10^{-1}	1.49×10^{-2}	4.52×10^{-2}	4.65×10^{-1}	5.25×10^{-1}
	Rail	5.68×10^{-2}	4.55×10^{-2}	1.06×10^{-3}	1.73×10^{-2}	6.39×10^{-2}
Portland, OR	Truck	2.25×10^{-1}	8.88×10^{-3}	3.72×10^{-2}	5.40×10^{-1}	5.86×10^{-1}
	Rail	5.94×10^{-2}	3.16×10^{-2}	1.05×10^{-3}	1.78×10^{-2}	5.04×10^{-2}
Tacoma, WA	Truck	2.26×10^{-1}	7.26×10^{-3}	3.44×10^{-2}	5.50×10^{-1}	5.92×10^{-1}
	Rail	5.95×10^{-2}	3.86×10^{-2}	1.17×10^{-3}	1.85×10^{-2}	5.83×10^{-2}
From DOE Sites/Canadian Border						
Alexandria Bay, NY	Truck	9.64×10^{-2}	4.11×10^{-3}	1.46×10^{-2}	1.92×10^{-1}	2.11×10^{-1}
	Rail	2.80×10^{-2}	3.17×10^{-2}	8.38×10^{-4}	9.07×10^{-3}	4.16×10^{-2}
Hanford Site	Truck	2.10×10^{-1}	7.29×10^{-2}	3.31×10^{-1}	5.10×10^{-1}	5.50×10^{-1}
	Rail	5.56×10^{-1}	1.64×10^{-1}	6.92×10^{-3}	1.60×10^{-1}	3.31×10^{-2}
Idaho National Engineering Laboratory	Truck	1.76×10^{-1}	6.45×10^{-2}	2.80×10^{-1}	4.24×10^{-1}	4.58×10^{-1}
	Rail	4.63×10^{-1}	1.41×10^{-1}	5.69×10^{-3}	1.33×10^{-1}	2.80×10^{-2}
Nevada Test Site	Truck	1.86×10^{-1}	7.63×10^{-2}	3.10×10^{-1}	4.45×10^{-1}	4.84×10^{-1}
	Rail	5.36×10^{-1}	1.68×10^{-1}	6.66×10^{-3}	1.63×10^{-1}	3.37×10^{-2}
Savannah River	Truck	4.23×10^{-1}	2.93×10^{-2}	7.96×10^{-2}	7.84×10^{-1}	8.93×10^{-1}
	Rail	1.87×10^{-1}	4.69×10^{-2}	2.38×10^{-3}	5.11×10^{-2}	1.00×10^{-2}
Sweetgrass, MT	Truck	1.94×10^{-1}	5.82×10^{-2}	2.93×10^{-1}	4.79×10^{-1}	5.14×10^{-1}
	Rail	4.70×10^{-1}	2.57×10^{-1}	8.41×10^{-3}	1.46×10^{-1}	4.11×10^{-2}

Shipments to Savannah River Site:						
Route(s)		***Crew***	***General Public***			
			Off-Link	***On-Link***	***Stops***	***Total***
From Eastern Ports						
Charleston, SC (NWS)	Truck	1.84×10^{-2}	7.44×10^{-4}	2.83×10^{-3}	3.89×10^{-2}	4.25×10^{-2}
	Rail	1.40×10^{-2}	1.77×10^{-3}	5.53×10^{-6}	4.77×10^{-3}	6.59×10^{-3}
Charleston, SC (Wando Terminal)	Truck	2.04×10^{-2}	1.10×10^{-3}	3.57×10^{-3}	4.21×10^{-2}	4.67×10^{-2}
	Rail	1.40×10^{-2}	1.77×10^{-3}	5.53×10^{-6}	4.77×10^{-3}	6.59×10^{-3}
Elizabeth, NJ	Truck	8.83×10^{-2}	5.31×10^{-3}	1.60×10^{-2}	1.70×10^{-1}	1.92×10^{-1}
	Rail	2.64×10^{-2}	4.14×10^{-2}	8.55×10^{-4}	8.08×10^{-3}	5.04×10^{-2}
Galveston, TX	Truck	1.02×10^{-1}	6.82×10^{-3}	1.94×10^{-2}	2.07×10^{-1}	2.33×10^{-1}
	Rail	3.15×10^{-2}	2.48×10^{-2}	7.41×10^{-4}	1.04×10^{-2}	3.59×10^{-2}
Jacksonville, FL	Truck	3.37×10^{-2}	7.25×10^{-4}	4.42×10^{-3}	7.81×10^{-2}	8.32×10^{-2}
	Rail	1.60×10^{-2}	3.06×10^{-3}	1.04×10^{-4}	4.61×10^{-3}	7.77×10^{-3}
Newport News, VA	Truck	5.24×10^{-2}	3.06×10^{-3}	9.51×10^{-3}	1.07×10^{-1}	1.20×10^{-1}
	Rail	2.18×10^{-2}	8.37×10^{-3}	3.55×10^{-4}	5.66×10^{-3}	1.44×10^{-2}
Norfolk, VA	Truck	4.89×10^{-2}	2.13×10^{-3}	7.66×10^{-3}	1.03×10^{-1}	1.13×10^{-1}
	Rail	2.06×10^{-2}	5.78×10^{-3}	2.66×10^{-4}	5.22×10^{-3}	1.13×10^{-2}

Shipments to Savannah River Site:							
Route(s)		*Crew*	*General Public*				
			Off-Link	*On-Link*	*Stops*	*Total*	
Philadelphia, PA	Truck	$8.21x10^{-2}$	$6.81x10^{-3}$	$1.78x10^{-2}$	$1.53x10^{-1}$	$1.78x10^{-1}$	
	Rail	$2.50x10^{-2}$	$2.67x10^{-2}$	$6.51x10^{-4}$	$7.21x10^{-3}$	$3.46x10^{-2}$	
Portsmouth, VA	Truck	$4.93x10^{-2}$	$2.21x10^{-3}$	$7.90x10^{-3}$	$1.04x10^{-1}$	$1.14x10^{-1}$	
	Rail	$2.01x10^{-2}$	$4.71x10^{-3}$	$2.39x10^{-4}$	$4.72x10^{-3}$	$9.67x10^{-3}$	
Savannah, GA	Truck	$2.29x10^{-2}$	$5.78x10^{-4}$	$3.04x10^{-3}$	$5.18x10^{-2}$	$5.54x10^{-2}$	
	Rail	$1.36x10^{-2}$	$8.37x10^{-4}$	$3.51x10^{-5}$	$4.39x10^{-3}$	$5.27x10^{-3}$	
MOTSU, NC	Truck	$2.22x10^{-2}$	$5.41x10^{-4}$	$3.06x10^{-3}$	$5.17x10^{-2}$	$5.53x10^{-2}$	
	Rail	$1.81x10^{-2}$	$3.82x10^{-3}$	$1.73x10^{-4}$	$4.51x10^{-3}$	$8.50x10^{-3}$	
Wilmington, NC	Truck	$2.96x10^{-2}$	$1.03x10^{-3}$	$4.30x10^{-3}$	$6.42x10^{-2}$	$6.95x10^{-2}$	
	Rail	$1.79x10^{-2}$	$3.71x10^{-3}$	$1.65x10^{-4}$	$4.46x10^{-3}$	$8.33x10^{-3}$	
From Western Ports							
NWS Concord, CA	Truck	$2.64x10^{-1}$	$1.83x10^{-2}$	$5.52x10^{-2}$	$5.76x10^{-1}$	$6.50x10^{-1}$	
	Rail	$6.54x10^{-2}$	$5.17x10^{-2}$	$1.52x10^{-3}$	$1.88x10^{-2}$	$7.20x10^{-2}$	
Long Beach, CA	Truck	$2.33x10^{-1}$	$1.61x10^{-2}$	$4.83x10^{-2}$	$5.06x10^{-1}$	$5.70x10^{-1}$	
	Rail	$6.63x10^{-2}$	$5.65x10^{-2}$	$1.51x10^{-3}$	$1.80x10^{-2}$	$7.60x10^{-2}$	
Portland, OR	Truck	$2.57x10^{-1}$	$1.17x10^{-2}$	$4.42x10^{-2}$	$5.96x10^{-1}$	$6.52x10^{-1}$	
	Rail	$6.49x10^{-2}$	$3.88x10^{-2}$	$1.34x10^{-3}$	$1.81x10^{-2}$	$5.82x10^{-2}$	
Seattle, WA	Truck	$2.54x10^{-1}$	$9.50x10^{-3}$	$4.00x10^{-2}$	$6.00x10^{-1}$	$6.50x10^{-1}$	
	Rail	$6.44x10^{-2}$	$4.13x10^{-2}$	$1.37x10^{-3}$	$1.82x10^{-2}$	$6.09x10^{-2}$	
Tacoma, WA	Truck	$2.58x10^{-1}$	$1.02x10^{-2}$	$4.15x10^{-2}$	$6.07x10^{-1}$	$6.59x10^{-1}$	
	Rail	$6.51x10^{-2}$	$4.59x10^{-2}$	$1.45x10^{-3}$	$1.87x10^{-2}$	$6.60x10^{-2}$	
From DOE Sites/Canadian Border							
Alexandria Bay, NY	Truck	$1.04x10^{-1}$	$4.58x10^{-3}$	$1.61x10^{-2}$	$2.10x10^{-1}$	$2.30x10^{-1}$	
	Rail	$3.33x10^{-2}$	$6.15x10^{-2}$	$1.30x10^{-3}$	$9.18x10^{-3}$	$7.20x10^{-2}$	
Hanford Site	Truck	$2.42x10^{-1}$	$1.02x10^{-2}$	$4.01x10^{-2}$	$5.65x10^{-1}$	$6.15x10^{-1}$	
	Rail	$6.15x10^{-2}$	$2.74x10^{-2}$	$1.08x10^{-3}$	$1.66x10^{-2}$	$4.51x10^{-2}$	
Idaho National Engineering Laboratory	Truck	$2.08x10^{-1}$	$9.34x10^{-3}$	$3.50x10^{-2}$	$4.79x10^{-1}$	$5.23x10^{-1}$	
	Rail	$5.23x10^{-2}$	$2.51x10^{-2}$	$9.56x10^{-4}$	$1.41x10^{-2}$	$4.02x10^{-2}$	
Nevada Test Site	Truck	$2.17x10^{-1}$	$1.05x10^{-2}$	$3.80x10^{-2}$	$5.00x10^{-1}$	$5.48x10^{-1}$	
	Rail	$5.96x10^{-2}$	$2.77x10^{-2}$	$1.05x10^{-3}$	$1.65x10^{-2}$	$4.53x10^{-2}$	
Oak Ridge Reservation	Truck	$4.23x10^{-2}$	$2.93x10^{-3}$	$7.96x10^{-3}$	$7.84x10^{-2}$	$8.93x10^{-2}$	
	Rail	$1.87x10^{-2}$	$4.69x10^{-3}$	$2.38x10^{-4}$	$5.11x10^{-3}$	$1.00x10^{-2}$	
Sweetgrass, MT	Truck	$2.26x10^{-1}$	$8.66x10^{-3}$	$3.62x10^{-2}$	$5.33x10^{-1}$	$5.78x10^{-1}$	
	Rail	$5.26x10^{-2}$	$3.30x10^{-2}$	$1.13x10^{-3}$	$1.49x10^{-2}$	$4.90x10^{-2}$	

[a] *Incident free risk factors are based on dose rates of 10 mrem per hr at 2 m (6.6 ft) (the regulatory limit).*

MOTSU = Military Ocean Terminal at Sunny Point, NWS = Naval Weapons Station

Table E-9 Accident Dose Risk per Shipment for All Spent Nuclear Fuel Types (Person-Rem/shipment)

Shipments to Hanford Site:		
Source/Route	*Truck*	*Rail*
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.51x10^{-4}$	$2.06x10^{-5}$
Charleston, SC (Wando Terminal)	$1.54x10^{-4}$	$2.06x10^{-5}$
Elizabeth, NJ	$1.30x10^{-4}$	$4.19x10^{-5}$
Galveston, TX	$9.53x10^{-5}$	$8.81x10^{-6}$
Jacksonville, FL	$1.71x10^{-4}$	$1.86x10^{-5}$

Shipments to Hanford Site:		
Source/Route	***Truck***	***Rail***
Newport News, VA	1.44×10^{-4}	3.28×10^{-5}
Norfolk, VA	1.47×10^{-4}	3.33×10^{-5}
Philadelphia, PA	1.31×10^{-4}	3.75×10^{-5}
Portsmouth, VA	1.49×10^{-4}	3.28×10^{-5}
Savannah, GA	1.65×10^{-4}	1.86×10^{-5}
MOTSU, NC	1.45×10^{-4}	2.14×10^{-5}
Wilmington, NC	1.51×10^{-4}	2.13×10^{-5}
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	6.54×10^{-5}	8.86×10^{-6}
Charleston, SC (Wando Terminal)	6.67×10^{-5}	8.86×10^{-6}
Elizabeth, NJ	5.59×10^{-5}	1.80×10^{-5}
Galveston, TX	4.12×10^{-5}	3.80×10^{-6}
Jacksonville, FL	7.39×10^{-5}	8.00×10^{-6}
Newport News, VA	6.21×10^{-5}	1.41×10^{-5}
Norfolk, VA	6.35×10^{-5}	1.43×10^{-5}
Philadelphia, PA	5.67×10^{-5}	1.61×10^{-5}
Portsmouth, VA	6.44×10^{-5}	1.41×10^{-5}
Savannah, GA	7.11×10^{-5}	8.03×10^{-6}
MOTSU, NC	6.28×10^{-5}	9.20×10^{-6}
Wilmington, NC	6.51×10^{-5}	9.15×10^{-6}
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	2.09×10^{-4}	2.86×10^{-5}
Charleston, SC (Wando Terminal)	2.13×10^{-4}	2.86×10^{-5}
Elizabeth, NJ	1.79×10^{-4}	5.83×10^{-5}
Galveston, TX	1.32×10^{-4}	1.22×10^{-5}
Jacksonville, FL	2.36×10^{-4}	2.58×10^{-5}
Newport News, VA	1.99×10^{-4}	4.56×10^{-5}
Norfolk, VA	2.03×10^{-4}	4.64×10^{-5}
Philadelphia, PA	1.82×10^{-4}	5.22×10^{-5}
Portsmouth, VA	2.06×10^{-4}	4.57×10^{-5}
Savannah, GA	2.27×10^{-4}	2.59×10^{-5}
MOTSU, NC	2.01×10^{-4}	2.97×10^{-5}
Wilmington, NC	2.08×10^{-4}	2.95×10^{-5}
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	4.33×10^{-4}	9.80×10^{-5}
Charleston, SC (Wando Terminal)	4.40×10^{-4}	9.80×10^{-5}
Elizabeth, NJ	4.49×10^{-4}	2.11×10^{-4}
Galveston, TX	2.68×10^{-4}	3.71×10^{-5}
Jacksonville, FL	4.81×10^{-4}	9.21×10^{-5}
Newport News, VA	4.23×10^{-4}	1.83×10^{-4}
Norfolk, VA	4.31×10^{-4}	1.85×10^{-4}
Philadelphia, PA	4.63×10^{-4}	1.97×10^{-4}
Portsmouth, VA	4.36×10^{-4}	1.84×10^{-4}
Savannah, GA	4.64×10^{-4}	9.25×10^{-5}
MOTSU, NC	4.27×10^{-4}	1.01×10^{-4}
Wilmington, NC	4.32×10^{-4}	1.01×10^{-4}
Calcined Target Material		
Charleston, SC (NWS)	3.96×10^{-2}	7.21×10^{-3}
Charleston, SC (Wando Terminal)	3.98×10^{-2}	7.21×10^{-3}
Elizabeth, NJ	6.73×10^{-2}	1.66×10^{-2}
Jacksonville, FL	4.09×10^{-2}	7.17×10^{-3}

Shipments to Hanford Site:		
Source/Route	**Truck**	**Rail**
Newport News, VA	4.22×10^{-2}	1.62×10^{-2}
Norfolk, VA	4.27×10^{-2}	1.63×10^{-2}
Philadelphia, PA	7.31×10^{-2}	1.62×10^{-2}
Portsmouth, VA	4.25×10^{-2}	1.63×10^{-2}
Savannah, GA	4.01×10^{-2}	7.21×10^{-3}
MOTSU, NC	4.35×10^{-2}	7.41×10^{-3}
Wilmington, NC	4.07×10^{-2}	7.40×10^{-3}
Galveston, TX	2.25×10^{-2}	2.16×10^{-3}
Oxidized Target Material		
Charleston, SC (NWS)	9.91×10^{-2}	1.80×10^{-2}
Charleston, SC (Wando Terminal)	9.95×10^{-2}	1.80×10^{-2}
Elizabeth, NJ	1.68×10^{-1}	4.14×10^{-2}
Jacksonville, FL	1.02×10^{-1}	1.79×10^{-2}
Newport News, VA	1.05×10^{-1}	4.06×10^{-2}
Norfolk, VA	1.07×10^{-1}	4.09×10^{-2}
Philadelphia, PA	1.83×10^{-1}	4.06×10^{-2}
Portsmouth, VA	1.06×10^{-1}	4.08×10^{-2}
Savannah, GA	1.00×10^{-1}	1.80×10^{-2}
MOTSU, NC	1.09×10^{-1}	1.85×10^{-2}
Wilmington, NC	1.02×10^{-1}	1.85×10^{-2}
Galveston, TX	5.62×10^{-2}	5.40×10^{-3}
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS, Concord CA	4.46×10^{-5}	1.50×10^{-5}
Long Beach, CA	7.36×10^{-5}	1.59×10^{-5}
Portland, OR	1.15×10^{-5}	2.10×10^{-6}
Tacoma, WA	8.55×10^{-6}	2.08×10^{-6}
RHF France Spent Nuclear Fuel		
NWS, Concord CA	1.93×10^{-5}	6.47×10^{-6}
Long Beach, CA	3.18×10^{-5}	6.86×10^{-6}
Portland, OR	4.99×10^{-6}	9.08×10^{-7}
Tacoma, WA	3.69×10^{-6}	8.99×10^{-7}
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	6.15×10^{-5}	2.07×10^{-5}
Long Beach, CA	1.02×10^{-4}	2.20×10^{-5}
Portland, OR	1.59×10^{-5}	2.91×10^{-6}
Tacoma, WA	1.18×10^{-5}	2.88×10^{-6}
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	1.15×10^{-4}	5.15×10^{-5}
Long Beach, CA	1.90×10^{-4}	5.50×10^{-5}
Portland, OR	2.97×10^{-5}	7.50×10^{-6}
Tacoma, WA	2.46×10^{-5}	7.79×10^{-6}
Calcined Target Material		
NWS Concord, CA	4.92×10^{-3}	1.50×10^{-3}
Long Beach, CA	7.06×10^{-3}	1.63×10^{-3}
Portland, OR	1.28×10^{-3}	2.54×10^{-4}
Tacoma, WA	1.90×10^{-3}	3.11×10^{-4}
Oxidized Target Material		
NWS Concord, CA	1.23×10^{-2}	3.75×10^{-3}
Long Beach, CA	1.77×10^{-2}	4.08×10^{-3}
Portland, OR	3.21×10^{-3}	6.36×10^{-4}

Shipments to Hanford Site:		
Source/Route	***Truck***	***Rail***
Tacoma, WA	4.75×10^{-3}	7.80×10^{-4}
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	1.52×10^{-4}	3.76×10^{-5}
Idaho National Engineering Laboratory	1.00×10^{-5}	3.08×10^{-6}
Nevada Test Site	5.07×10^{-5}	4.62×10^{-6}
Oak Ridge Reservation	1.13×10^{-4}	1.16×10^{-5}
Savannah River	1.61×10^{-4}	1.97×10^{-5}
Sweetgrass, MT	2.25×10^{-5}	1.56×10^{-6}
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	6.56×10^{-5}	1.62×10^{-5}
Idaho National Engineering Laboratory	4.33×10^{-6}	1.33×10^{-6}
Nevada Test Site	2.19×10^{-5}	1.99×10^{-6}
Oak Ridge Reservation	4.86×10^{-5}	4.97×10^{-6}
Savannah River	6.94×10^{-5}	8.47×10^{-6}
Sweetgrass, MT	9.72×10^{-6}	6.72×10^{-7}
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	2.10×10^{-4}	5.23×10^{-5}
Idaho National Engineering Laboratory	1.38×10^{-5}	4.27×10^{-6}
Nevada Test Site	6.99×10^{-5}	6.39×10^{-6}
Oak Ridge Reservation	1.56×10^{-4}	1.62×10^{-5}
Savannah River	2.22×10^{-4}	2.73×10^{-5}
Sweetgrass, MT	3.11×10^{-5}	2.16×10^{-6}
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	4.99×10^{-4}	1.98×10^{-4}
Idaho National Engineering Laboratory	2.94×10^{-5}	1.21×10^{-5}
Nevada Test Site	1.29×10^{-4}	1.74×10^{-5}
Oak Ridge Reservation	3.35×10^{-4}	7.04×10^{-5}
Savannah River	4.53×10^{-4}	9.49×10^{-5}
Sweetgrass, MT	6.49×10^{-5}	6.18×10^{-6}
Calcined Target Material		
Alexandria Bay, NY	6.72×10^{-2}	1.63×10^{-2}
Idaho National Engineering Laboratory	2.69×10^{-3}	5.83×10^{-4}
Nevada Test Site	4.74×10^{-3}	7.41×10^{-4}
Oak Ridge Reservation	3.55×10^{-2}	6.78×10^{-3}
Savannah River	3.96×10^{-2}	7.11×10^{-3}
Sweetgrass, MT	6.01×10^{-3}	3.03×10^{-4}
Oxidized Target Material		
Alexandria Bay, NY	1.68×10^{-1}	4.07×10^{-2}
Idaho National Engineering Laboratory	6.74×10^{-3}	1.46×10^{-3}
Nevada Test Site	1.19×10^{-2}	1.85×10^{-3}
Oak Ridge Reservation	8.88×10^{-2}	1.70×10^{-2}
Savannah River	9.90×10^{-2}	1.78×10^{-2}
Sweetgrass, MT	1.50×10^{-2}	7.57×10^{-4}

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	1.45×10^{-4}	1.85×10^{-5}
Charleston, SC (Wando Terminal)	1.48×10^{-4}	1.85×10^{-5}
Elizabeth, NJ	1.23×10^{-4}	3.98×10^{-5}

Shipments to Idaho National Engineering Laboratory:		
Source/Route	Truck	Rail
Galveston, TX	$8.88x10^{-5}$	$6.71x10^{-6}$
Jacksonville, FL	$1.62x10^{-4}$	$1.65x10^{-5}$
Newport News, VA	$1.37x10^{-4}$	$3.07x10^{-5}$
Norfolk, VA	$1.39x10^{-4}$	$3.12x10^{-5}$
Philadelphia, PA	$1.25x10^{-4}$	$3.54x10^{-5}$
Portsmouth, VA	$1.43x10^{-4}$	$3.07x10^{-5}$
Savannah, GA	$1.58x10^{-4}$	$1.66x10^{-5}$
MOTSU, NC	$1.34x10^{-4}$	$1.93x10^{-5}$
Wilmington, NC	$1.44x10^{-4}$	$1.92x10^{-5}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$6.26x10^{-5}$	$7.96x10^{-6}$
Charleston, SC (Wando Terminal)	$6.39x10^{-5}$	$7.96x10^{-6}$
Elizabeth, NJ	$5.31x10^{-5}$	$1.71x10^{-5}$
Galveston, TX	$3.84x10^{-5}$	$2.89x10^{-6}$
Jacksonville, FL	$7.02x10^{-5}$	$7.10x10^{-6}$
Newport News, VA	$5.93x10^{-5}$	$1.32x10^{-5}$
Norfolk, VA	$6.01x10^{-5}$	$1.34x10^{-5}$
Philadelphia, PA	$5.39x10^{-5}$	$1.52x10^{-5}$
Portsmouth, VA	$6.16x10^{-5}$	$1.32x10^{-5}$
Savannah, GA	$6.83x10^{-5}$	$7.12x10^{-6}$
MOTSU, NC	$5.78x10^{-5}$	$8.30x10^{-6}$
Wilmington, NC	$6.22x10^{-5}$	$8.25x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$2.00x10^{-4}$	$2.57x10^{-5}$
Charleston, SC (Wando Terminal)	$2.04x10^{-4}$	$2.57x10^{-5}$
Elizabeth, NJ	$1.23x10^{-4}$	$5.54x10^{-5}$
Galveston, TX	$1.70x10^{-4}$	$9.29x10^{-6}$
Jacksonville, FL	$2.24x10^{-4}$	$2.29x10^{-5}$
Newport News, VA	$1.90x10^{-4}$	$4.27x10^{-5}$
Norfolk, VA	$1.92x10^{-4}$	$4.35x10^{-5}$
Portsmouth, VA	$1.97x10^{-4}$	$4.28x10^{-5}$
Savannah, GA	$2.18x10^{-4}$	$2.30x10^{-5}$
MOTSU, NC	$1.85x10^{-4}$	$2.68x10^{-5}$
Wilmington, NC	$1.99x10^{-4}$	$2.66x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.14x10^{-4}$	$8.94x10^{-5}$
Charleston, SC (Wando Terminal)	$4.21x10^{-4}$	$8.94x10^{-5}$
Elizabeth, NJ	$4.30x10^{-4}$	$2.02x10^{-4}$
Galveston, TX	$4.49x10^{-4}$	$8.47x10^{-5}$
Jacksonville, FL	$4.57x10^{-4}$	$8.36x10^{-5}$
Newport News, VA	$4.04x10^{-4}$	$1.75x10^{-4}$
Norfolk, VA	$4.09x10^{-4}$	$1.77x10^{-4}$
Philadelphia, PA	$4.44x10^{-4}$	$1.88x10^{-4}$
Portsmouth, VA	$4.17x10^{-4}$	$1.75x10^{-4}$
Savannah, GA	$4.45x10^{-4}$	$8.39x10^{-5}$
MOTSU, NC	$3.97x10^{-4}$	$9.26x10^{-5}$
Wilmington, NC	$4.13x10^{-4}$	$9.22x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$3.79x10^{-2}$	$6.76x10^{-3}$
Charleston, SC (Wando Terminal)	$3.81x10^{-2}$	$6.76x10^{-3}$
Elizabeth, NJ	$6.57x10^{-2}$	$1.61x10^{-2}$

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
Jacksonville, FL	$3.91x10^{-2}$	$6.72x10^{-3}$
Newport News, VA	$4.05x10^{-2}$	$1.58x10^{-2}$
Norfolk, VA	$4.10x10^{-2}$	$1.59x10^{-2}$
Philadelphia, PA	$7.15x10^{-2}$	$1.58x10^{-2}$
Portsmouth, VA	$4.09x10^{-2}$	$1.59x10^{-2}$
Savannah, GA	$3.84x10^{-2}$	$6.76x10^{-3}$
MOTSU, NC	$4.18x10^{-2}$	$6.96x10^{-3}$
Wilmington, NC	$3.91x10^{-2}$	$6.95x10^{-3}$
Galveston, TX	$2.08x10^{-2}$	$1.71x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$9.49x10^{-2}$	$1.69x10^{-2}$
Charleston, SC (Wando Terminal)	$9.53x10^{-2}$	$1.69x10^{-2}$
Elizabeth, NJ	$1.64x10^{-1}$	$4.03x10^{-2}$
Jacksonville, FL	$9.79x10^{-2}$	$1.68x10^{-2}$
Newport News, VA	$1.01x10^{-1}$	$3.95x10^{-2}$
Norfolk, VA	$1.03x10^{-1}$	$3.98x10^{-2}$
Philadelphia, PA	$1.79x10^{-1}$	$3.95x10^{-2}$
Portsmouth, VA	$1.02x10^{-1}$	$3.96x10^{-2}$
Savannah, GA	$9.62x10^{-2}$	$1.69x10^{-2}$
MOTSU, NC	$1.05x10^{-1}$	$1.74x10^{-2}$
Wilmington, NC	$9.78x10^{-2}$	$1.74x10^{-2}$
Galveston, TX	$5.21x10^{-2}$	$4.28x10^{-3}$
	From Western Ports	
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$3.83x10^{-5}$	$4.00x10^{-6}$
Long Beach, CA	$7.40x10^{-5}$	$1.38x10^{-5}$
Portland, OR	$1.93x10^{-5}$	$4.94x10^{-6}$
Tacoma, WA	$1.66x10^{-5}$	$6.85x10^{-6}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$1.66x10^{-5}$	$1.73x10^{-6}$
Long Beach, CA	$3.20x10^{-5}$	$5.96x10^{-6}$
Portland, OR	$8.35x10^{-6}$	$2.13x10^{-6}$
Tacoma, WA	$7.19x10^{-6}$	$2.95x10^{-6}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$5.29x10^{-5}$	$5.53x10^{-6}$
Long Beach, CA	$1.02x10^{-4}$	$1.91x10^{-5}$
Portland, OR	$2.67x10^{-5}$	$6.84x10^{-6}$
Tacoma, WA	$2.30x10^{-5}$	$9.47x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$9.68x10^{-5}$	$1.42x10^{-5}$
Long Beach, CA	$1.85x10^{-4}$	$4.64x10^{-5}$
Portland, OR	$5.19x10^{-5}$	$1.85x10^{-5}$
Tacoma, WA	$4.88x10^{-5}$	$2.50x10^{-5}$
Calcined Target Material		
NWS Concord, CA	$3.21x10^{-3}$	$4.70x10^{-4}$
Long Beach, CA	$4.75x10^{-3}$	$1.18x10^{-3}$
Portland, OR	$3.23x10^{-3}$	$7.61x10^{-4}$
Tacoma, WA	$4.35x10^{-3}$	$9.55x10^{-4}$
Oxidized Target Material		
NWS Concord, CA	$8.05x10^{-3}$	$1.18x10^{-3}$
Long Beach, CA	$1.19x10^{-2}$	$2.95x10^{-3}$

Shipments to Idaho National Engineering Laboratory:		
Source/Route	***Truck***	***Rail***
Portland, OR	$8.07x10^{-3}$	$1.91x10^{-3}$
Tacoma, WA	$1.09x10^{-2}$	$2.39x10^{-3}$
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$1.45x10^{-4}$	$3.55x10^{-5}$
Hanford Site	$1.00x10^{-5}$	$3.08x10^{-6}$
Nevada Test Site	$4.41x10^{-5}$	$2.52x10^{-6}$
Oak Ridge Reservation	$1.06x10^{-4}$	$9.48x10^{-6}$
Savannah River	$1.54x10^{-4}$	$1.76x10^{-5}$
Sweetgrass, MT	$1.59x10^{-5}$	$3.84x10^{-6}$
RHF France Spent Nuclear Fuel		
Hanford Site	$4.33x10^{-6}$	$1.33x10^{-6}$
Nevada Test Site	$1.91x10^{-5}$	$1.09x10^{-6}$
Oak Ridge Reservation	$4.58x10^{-5}$	$4.07x10^{-6}$
Savannah River	$6.66x10^{-5}$	$7.57x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$2.01x10^{-4}$	$4.94x10^{-5}$
Hanford Site	$1.38x10^{-5}$	$4.27x10^{-6}$
Nevada Test Site	$6.09x10^{-5}$	$3.48x10^{-6}$
Oak Ridge Reservation	$1.46x10^{-4}$	$1.33x10^{-5}$
Savannah River	$2.13x10^{-4}$	$2.44x10^{-5}$
Sweetgrass, MT	$2.20x10^{-5}$	$5.32x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Hanford Site	$2.94x10^{-5}$	$1.21x10^{-5}$
Nevada Test Site	$1.09x10^{-4}$	$8.79x10^{-6}$
Oak Ridge Reservation	$3.16x10^{-4}$	$6.19x10^{-5}$
Savannah River	$4.34x10^{-4}$	$8.63x10^{-5}$
Calcined Target Material		
Alexandria Bay, NY	$6.55x10^{-2}$	$1.58x10^{-2}$
Hanford Site	$2.69x10^{-3}$	$5.83x10^{-4}$
Nevada Test Site	$3.10x10^{-3}$	$2.90x10^{-4}$
Oak Ridge Reservation	$3.38x10^{-2}$	$6.33x10^{-3}$
Savannah River	$3.79x10^{-2}$	$6.66x10^{-3}$
Sweetgrass, MT	$3.56x10^{-3}$	$8.19x10^{-4}$
Oxidized Target Material		
Alexandria Bay, NY	$1.64x10^{-1}$	$3.96x10^{-2}$
Hanford	$6.74x10^{-3}$	$1.46x10^{-3}$
Nevada Test Site	$7.76x10^{-3}$	$7.27x10^{-4}$
Oak Ridge Reservation	$8.47x10^{-2}$	$1.58x10^{-2}$
Savannah River	$9.49x10^{-2}$	$1.67x10^{-2}$
Sweetgrass, MT	$8.91x10^{-3}$	$2.05x10^{-3}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.65x10^{-4}$	$1.99x10^{-5}$
Charleston, SC (Wando Terminal)	$1.68x10^{-4}$	$1.99x10^{-5}$
Elizabeth, NJ	$1.77x10^{-4}$	$4.12x10^{-5}$
Galveston, TX	$9.13x10^{-5}$	$5.16x10^{-6}$
Jacksonville, FL	$1.89x10^{-4}$	$1.79x10^{-5}$
Newport News, VA	$1.64x10^{-4}$	$3.21x10^{-5}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Norfolk, VA	1.66×10^{-4}	3.27×10^{-5}
Philadelphia, PA	1.75×10^{-4}	3.68×10^{-5}
Portsmouth, VA	1.69×10^{-4}	3.22×10^{-5}
Savannah, GA	1.85×10^{-4}	1.80×10^{-5}
MOTSU, NC	1.65×10^{-4}	2.07×10^{-5}
Wilmington, NC	1.71×10^{-4}	2.06×10^{-5}
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	7.12×10^{-5}	8.57×10^{-6}
Charleston, SC (Wando Terminal)	7.25×10^{-5}	8.57×10^{-6}
Elizabeth, NJ	7.66×10^{-5}	1.78×10^{-5}
Galveston, TX	3.95×10^{-5}	2.42×10^{-6}
Jacksonville, FL	8.17×10^{-5}	7.71×10^{-6}
Newport News, VA	7.08×10^{-5}	1.38×10^{-5}
Norfolk, VA	7.16×10^{-5}	1.40×10^{-5}
Philadelphia, PA	7.56×10^{-5}	1.58×10^{-5}
Portsmouth, VA	7.31×10^{-5}	1.38×10^{-5}
Savannah, GA	7.98×10^{-5}	7.74×10^{-6}
MOTSU, NC	7.14×10^{-5}	8.91×10^{-6}
Wilmington, NC	7.37×10^{-5}	8.86×10^{-6}
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	2.27×10^{-4}	2.76×10^{-5}
Charleston, SC (Wando Terminal)	2.32×10^{-4}	2.76×10^{-5}
Elizabeth, NJ	2.45×10^{-4}	5.73×10^{-5}
Galveston, TX	1.26×10^{-4}	7.77×10^{-6}
Jacksonville, FL	2.61×10^{-4}	2.49×10^{-5}
Newport News, VA	2.26×10^{-4}	4.47×10^{-5}
Norfolk, VA	2.29×10^{-4}	4.55×10^{-5}
Philadelphia, PA	2.42×10^{-4}	5.12×10^{-5}
Portsmouth, VA	2.34×10^{-4}	4.48×10^{-5}
Savannah, GA	2.55×10^{-4}	2.50×10^{-5}
MOTSU, NC	2.28×10^{-4}	2.87×10^{-5}
Wilmington, NC	2.36×10^{-4}	2.86×10^{-5}
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	4.62×10^{-4}	9.35×10^{-5}
Charleston, SC (Wando Terminal)	4.69×10^{-4}	9.35×10^{-5}
Elizabeth, NJ	5.19×10^{-4}	2.07×10^{-4}
Galveston, TX	2.37×10^{-4}	2.44×10^{-5}
Jacksonville, FL	5.21×10^{-4}	8.77×10^{-5}
Newport News, VA	4.68×10^{-4}	1.79×10^{-4}
Norfolk, VA	4.73×10^{-4}	1.81×10^{-4}
Philadelphia, PA	5.17×10^{-4}	1.92×10^{-4}
Portsmouth, VA	4.81×10^{-4}	1.79×10^{-4}
Savannah, GA	5.10×10^{-4}	8.80×10^{-5}
MOTSU, NC	4.19×10^{-4}	9.67×10^{-5}
Wilmington, NC	4.78×10^{-4}	9.63×10^{-5}
20 yr old vitrified HLW		
Charleston, SC (NWS)	1.40×10^{-3}	2.33×10^{-4}
Charleston, SC (Wando Terminal)	1.42×10^{-3}	2.33×10^{-4}
Elizabeth, NJ	1.70×10^{-3}	5.33×10^{-4}
Jacksonville, FL	1.45×10^{-3}	2.27×10^{-4}
Newport News, VA	1.45×10^{-3}	5.06×10^{-4}

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Norfolk, VA	$1.47x10^{-3}$	$5.10x10^{-4}$
Philadelphia, PA	$1.73x10^{-3}$	$5.14x10^{-4}$
Portsmouth, VA	$1.46x10^{-3}$	$5.07x10^{-4}$
Savannah, GA	$1.42x10^{-3}$	$2.28x10^{-4}$
MOTSU, NC	$7.07x10^{-4}$	$2.40x10^{-4}$
Wilmington, NC	$1.44x10^{-3}$	$2.40x10^{-4}$
Galveston, TX	$4.20x10^{-4}$	$4.70x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$3.90x10^{-2}$	$6.78x10^{-3}$
Charleston, SC (Wando Terminal)	$3.92x10^{-2}$	$6.78x10^{-3}$
Elizabeth, NJ	$4.86x10^{-2}$	$1.61x10^{-2}$
Jacksonville, FL	$4.03x10^{-2}$	$6.74x10^{-3}$
Newport News, VA	$4.16x10^{-2}$	$1.58x10^{-2}$
Norfolk, VA	$4.21x10^{-2}$	$1.59x10^{-2}$
Philadelphia, PA	$5.03x10^{-2}$	$1.58x10^{-2}$
Portsmouth, VA	$4.20x10^{-2}$	$1.59x10^{-2}$
Savannah, GA	$3.96x10^{-2}$	$6.78x10^{-3}$
MOTSU, NC	$1.79x10^{-2}$	$6.98x10^{-3}$
Wilmington, NC	$4.02x10^{-2}$	$6.97x10^{-3}$
Galveston, TX	$1.11x10^{-2}$	$1.43x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$9.76x10^{-2}$	$1.70x10^{-2}$
Charleston, SC (Wando Terminal)	$9.81x10^{-2}$	$1.70x10^{-2}$
Elizabeth, NJ	$1.22x10^{-1}$	$4.04x10^{-2}$
Jacksonville, FL	$1.01x10^{-1}$	$1.69x10^{-2}$
Newport News, VA	$1.04x10^{-1}$	$3.95x10^{-2}$
Norfolk, VA	$1.05x10^{-1}$	$3.98x10^{-2}$
Philadelphia, PA	$1.26x10^{-1}$	$3.95x10^{-2}$
Portsmouth, VA	$1.05x10^{-1}$	$3.97x10^{-2}$
Savannah, GA	$9.90x10^{-2}$	$1.70x10^{-2}$
MOTSU, NC	$4.48x10^{-2}$	$1.75x10^{-2}$
Wilmington, NC	$1.01x10^{-1}$	$1.74x10^{-2}$
Galveston, TX	$2.78x10^{-2}$	$3.57x10^{-3}$
From Western Ports		
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$4.88x10^{-5}$	$8.09x10^{-6}$
Long Beach, CA	$5.28x10^{-5}$	$1.13x10^{-5}$
Portland, OR	$6.10x10^{-5}$	$6.48x10^{-6}$
Tacoma, WA	$5.73x10^{-5}$	$8.38x10^{-6}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$2.11x10^{-5}$	$3.49x10^{-6}$
Long Beach, CA	$2.28x10^{-5}$	$4.89x10^{-6}$
Portland, OR	$2.64x10^{-5}$	$2.80x10^{-6}$
Tacoma, WA	$2.48x10^{-5}$	$3.62x10^{-6}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$6.73x10^{-5}$	$1.12x10^{-5}$
Long Beach, CA	$7.28x10^{-5}$	$1.56x10^{-5}$
Portland, OR	$8.41x10^{-5}$	$8.96x10^{-6}$
Tacoma, WA	$7.91x10^{-5}$	$1.16x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$1.26x10^{-4}$	$2.91x10^{-5}$

Shipments to Nevada Test Site:		
Source/Route	***Truck***	***Rail***
Long Beach, CA	$1.31x10^{-4}$	$3.77x10^{-5}$
Portland, OR	$1.53x10^{-4}$	$2.37x10^{-5}$
Tacoma, WA	$1.48x10^{-4}$	$3.02x10^{-5}$
Calcined Target Material		
NWS Concord, CA	$4.13x10^{-3}$	$9.39x10^{-4}$
Long Beach, CA	$2.42x10^{-3}$	$8.94x10^{-4}$
Portland, OR	$5.26x10^{-3}$	$9.19x10^{-4}$
Tacoma, WA	$6.39x10^{-3}$	$1.11x10^{-3}$
Oxidized Target Material		
NWS Concord, CA	$1.03x10^{-2}$	$2.35x10^{-3}$
Long Beach, CA	$6.05x10^{-3}$	$2.24x10^{-3}$
Portland, OR	$1.32x10^{-2}$	$2.30x10^{-3}$
Tacoma, WA	$1.60x10^{-2}$	$2.79x10^{-3}$
From DOE Sites/Canadian Border		
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$1.79x10^{-4}$	$3.69x10^{-5}$
Hanford Site	$5.07x10^{-5}$	$4.62x10^{-6}$
Idaho National Engineering Laboratory	$4.41x10^{-5}$	$2.52x10^{-6}$
Oak Ridge Reservation	$1.33x10^{-4}$	$1.09x10^{-5}$
Savannah River	$1.81x10^{-4}$	$1.90x10^{-5}$
Sweetgrass, MT	$6.08x10^{-5}$	$5.38x10^{-6}$
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	$7.71x10^{-5}$	$1.59x10^{-5}$
Hanford Site	$2.19x10^{-5}$	$1.99x10^{-6}$
Idaho National Engineering Laboratory	$1.91x10^{-5}$	$1.09x10^{-6}$
Oak Ridge Reservation	$5.73x10^{-5}$	$4.68x10^{-6}$
Savannah River	$7.81x10^{-5}$	$8.18x10^{-6}$
Sweetgrass, MT	$2.63x10^{-5}$	$2.32x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$2.47x10^{-4}$	$5.14x10^{-5}$
Hanford Site	$6.99x10^{-5}$	$6.39x10^{-6}$
Idaho National Engineering Laboratory	$6.09x10^{-5}$	$3.48x10^{-6}$
Oak Ridge Reservation	$1.83x10^{-4}$	$1.52x10^{-5}$
Savannah River	$2.49x10^{-4}$	$2.64x10^{-5}$
Sweetgrass, MT	$8.38x10^{-5}$	$7.45x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	$5.58x10^{-4}$	$1.93x10^{-4}$
Hanford Site	$1.29x10^{-4}$	$1.74x10^{-5}$
Idaho National Engineering Laboratory	$1.09x10^{-4}$	$8.79x10^{-6}$
Oak Ridge Reservation	$3.81x10^{-4}$	$6.60x10^{-5}$
Savannah River	$4.99x10^{-4}$	$9.04x10^{-5}$
Sweetgrass MT	$1.54x10^{-4}$	$2.09x10^{-5}$
Calcined Target Material		
Alexandria Bay, NY	$6.55x10^{-2}$	$1.58x10^{-2}$
Hanford	$4.74x10^{-3}$	$7.41x10^{-4}$
Idaho National Engineering Laboratory	$3.10x10^{-3}$	$2.90x10^{-4}$
Oak Ridge Reservation	$3.50x10^{-2}$	$6.35x10^{-3}$
Savannah River	$3.91x10^{-2}$	$6.68x10^{-3}$
Sweetgrass, MT	$6.29x10^{-3}$	$9.78x10^{-4}$
Oxidized Target Material		
Alexandria Bay, NY	$1.64x10^{-1}$	$3.96x10^{-2}$

Shipments to Nevada Test Site:		
Source/Route	**Truck**	**Rail**
Hanford	$1.19x10^{-2}$	$1.85x10^{-3}$
Idaho National Engineering Laboratory	$7.76x10^{-3}$	$7.27x10^{-4}$
Oak Ridge Reservation	$8.75x10^{-2}$	$1.59x10^{-2}$
Savannah River	$9.77x10^{-2}$	$1.67x10^{-2}$
Sweetgrass, MT	$1.58x10^{-2}$	$2.45x10^{-3}$

Shipments to Oak Ridge Reservation:		
Source/Route	**Truck**	**Rail**
From Eastern Ports		
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$3.28x10^{-5}$	$5.39x10^{-6}$
Charleston, SC (Wando Terminal)	$3.59x10^{-5}$	$5.39x10^{-6}$
Elizabeth, NJ	$5.86x10^{-5}$	$2.60x10^{-5}$
Galveston, TX	$6.99x10^{-5}$	$1.77x10^{-5}$
Jacksonville, FL	$4.98x10^{-5}$	$6.73x10^{-6}$
Newport News, VA	$4.20x10^{-5}$	$7.94x10^{-6}$
Norfolk, VA	$3.90x10^{-5}$	$6.68x10^{-6}$
Philadelphia, PA	$6.36x10^{-5}$	$1.76x10^{-5}$
Portsmouth, VA	$4.74x10^{-5}$	$6.18x10^{-6}$
Savannah, GA	$3.20x10^{-5}$	$6.79x10^{-6}$
MOTSU, NC	$3.60x10^{-5}$	$5.33x10^{-6}$
Wilmington, NC	$3.85x10^{-5}$	$5.22x10^{-6}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.42x10^{-5}$	$2.33x10^{-6}$
Charleston, SC (Wando Terminal)	$1.55x10^{-5}$	$2.33x10^{-6}$
Elizabeth, NJ	$2.53x10^{-5}$	$1.13x10^{-5}$
Galveston, TX	$3.02x10^{-5}$	$7.64x10^{-6}$
Jacksonville, FL	$2.15x10^{-5}$	$2.91x10^{-6}$
Newport News, VA	$1.81x10^{-5}$	$3.43x10^{-6}$
Norfolk, VA	$1.69x10^{-5}$	$2.88x10^{-6}$
Philadelphia, PA	$2.75x10^{-5}$	$7.60x10^{-6}$
Portsmouth, VA	$2.05x10^{-5}$	$2.67x10^{-6}$
Savannah, GA	$1.38x10^{-5}$	$2.93x10^{-6}$
MOTSU, NC	$1.56x10^{-5}$	$2.30x10^{-6}$
Wilmington, NC	$1.66x10^{-5}$	$2.25x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$4.53x10^{-5}$	$7.45x10^{-6}$
Charleston, SC (Wando Terminal)	$4.95x10^{-5}$	$7.45x10^{-6}$
Elizabeth, NJ	$8.08x10^{-5}$	$3.60x10^{-5}$
Galveston, TX	$9.63x10^{-5}$	$2.44x10^{-5}$
Jacksonville, FL	$6.86x10^{-5}$	$9.30x10^{-6}$
Newport News, VA	$5.79x10^{-5}$	$1.10x10^{-5}$
Norfolk, VA	$5.38x10^{-5}$	$9.23x10^{-6}$
Philadelphia, PA	$8.77x10^{-5}$	$2.43x10^{-5}$
Portsmouth, VA	$6.53x10^{-5}$	$8.54x10^{-6}$
Savannah, GA	$4.41x10^{-5}$	$9.38x10^{-6}$
MOTSU, NC	$4.97x10^{-5}$	$7.36x10^{-6}$
Wilmington, NC	$5.31x10^{-5}$	$7.21x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$8.32x10^{-5}$	$1.94x10^{-5}$
Charleston, SC (Wando Terminal)	$9.06x10^{-5}$	$1.94x10^{-5}$
Elizabeth, NJ	$1.53x10^{-4}$	$8.64x10^{-5}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
Galveston, TX	$1.75x10^{-4}$	$6.15x10^{-5}$
Jacksonville, FL	$1.26x10^{-4}$	$2.40x10^{-5}$
Newport News, VA	$1.09x10^{-4}$	$2.84x10^{-5}$
Norfolk, VA	$1.01x10^{-4}$	$2.43x10^{-5}$
Philadelphia, PA	$1.66x10^{-4}$	$5.90x10^{-5}$
Portsmouth, VA	$1.22x10^{-4}$	$2.26x10^{-5}$
Savannah, GA	$8.13x10^{-5}$	$2.43x10^{-5}$
MOTSU, NC	$9.18x10^{-5}$	$1.94x10^{-5}$
Wilmington, NC	$9.85x10^{-5}$	$1.90x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$4.14x10^{-3}$	$7.70x10^{-4}$
Charleston, SC (Wando Terminal)	$4.32x10^{-3}$	$7.70x10^{-4}$
Elizabeth, NJ	$8.03x10^{-3}$	$2.18x10^{-3}$
Jacksonville, FL	$5.87x10^{-3}$	$9.16x10^{-4}$
Newport News, VA	$5.46x10^{-3}$	$1.10x10^{-3}$
Norfolk, VA	$5.50x10^{-3}$	$9.94x10^{-4}$
Philadelphia, PA	$8.57x10^{-3}$	$1.59x10^{-3}$
Portsmouth, VA	$5.79x10^{-3}$	$9.45x10^{-4}$
Savannah, GA	$4.19x10^{-3}$	$9.55x10^{-4}$
MOTSU, NC	$4.78x10^{-3}$	$8.03x10^{-4}$
Wilmington, NC	$5.32x10^{-3}$	$7.89x10^{-4}$
Galveston, TX	$7.44x10^{-3}$	$2.10x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$1.04x10^{-2}$	$1.93x10^{-3}$
Charleston, SC (Wando Terminal)	$1.08x10^{-2}$	$1.93x10^{-3}$
Elizabeth, NJ	$2.01x10^{-2}$	$5.46x10^{-3}$
Jacksonville, FL	$1.47x10^{-2}$	$2.30x10^{-3}$
Newport News, VA	$1.37x10^{-2}$	$2.76x10^{-3}$
Norfolk, VA	$1.38x10^{-2}$	$2.49x10^{-3}$
Philadelphia, PA	$2.15x10^{-2}$	$3.97x10^{-3}$
Portsmouth, VA	$1.45x10^{-2}$	$2.37x10^{-3}$
Savannah, GA	$1.05x10^{-2}$	$2.39x10^{-3}$
MOTSU, NC	$1.20x10^{-2}$	$2.01x10^{-3}$
Wilmington, NC	$1.33x10^{-2}$	$1.98x10^{-3}$
Galveston, TX	$1.86x10^{-2}$	$5.25x10^{-3}$
	From Western Ports	
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	$1.46x10^{-4}$	$1.56x10^{-5}$
Long Beach, CA	$1.50x10^{-4}$	$2.08x10^{-5}$
Portland, OR	$1.33x10^{-4}$	$2.40x10^{-5}$
Tacoma, WA	$1.20x10^{-4}$	$2.55x10^{-5}$
RHF France Spent Nuclear Fuel		
NWS Concord, CA	$6.33x10^{-5}$	$6.72x10^{-6}$
Long Beach, CA	$6.50x10^{-5}$	$8.97x10^{-6}$
Portland, OR	$5.75x10^{-5}$	$1.03x10^{-5}$
Tacoma, WA	$5.16x10^{-5}$	$1.10x10^{-5}$
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	$2.02x10^{-4}$	$2.17x10^{-5}$
Long Beach, CA	$2.07x10^{-4}$	$2.89x10^{-5}$
Portland, OR	$1.84x10^{-4}$	$3.35x10^{-5}$
Tacoma, WA	$1.65x10^{-4}$	$3.56x10^{-5}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	$3.81x10^{-4}$	$7.68x10^{-5}$
Long Beach, CA	$3.86x10^{-4}$	$9.42x10^{-5}$
Portland, OR	$3.85x10^{-4}$	$1.25x10^{-4}$
Tacoma, WA	$3.78x10^{-4}$	$1.30x10^{-4}$
Calcined Target Material		
NWS Concord, CA	$1.77x10^{-2}$	$5.82x10^{-3}$
Long Beach, CA	$1.60x10^{-2}$	$6.15x10^{-3}$
Portland, OR	$3.65x10^{-2}$	$8.50x10^{-3}$
Tacoma, WA	$4.74x10^{-2}$	$8.59x10^{-3}$
Oxidized Target Material		
NWS Concord, CA	$4.43x10^{-2}$	$1.46x10^{-2}$
Long Beach, CA	$4.01x10^{-2}$	$1.54x10^{-2}$
Portland, OR	$9.13x10^{-2}$	$2.13x10^{-2}$
Tacoma, WA	$1.19x10^{-1}$	$2.15x10^{-2}$
	From DOE Sites/Canadian Border	
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	$6.57x10^{-5}$	$1.83x10^{-5}$
Hanford Site	$1.13x10^{-4}$	$1.16x10^{-5}$
Idaho National Engineering Laboratory	$1.06x10^{-4}$	$9.48x10^{-6}$
Nevada Test Site	$1.33x10^{-4}$	$1.09x10^{-5}$
Savannah River	$5.10x10^{-5}$	$3.67x10^{-6}$
Sweetgrass, MT	$1.21x10^{-4}$	$2.12x10^{-5}$
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	$2.84x10^{-5}$	$7.91x10^{-6}$
Hanford Site	$4.86x10^{-5}$	$4.97x10^{-6}$
Idaho National Engineering Laboratory	$4.58x10^{-5}$	$4.07x10^{-6}$
Nevada Test Site	$5.73x10^{-5}$	$4.68x10^{-6}$
Savannah River	$2.21x10^{-5}$	$1.59x10^{-6}$
Sweetgrass, MT	$5.22x10^{-5}$	$9.11x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	$9.07x10^{-5}$	$2.53x10^{-5}$
Hanford Site	$1.56x10^{-4}$	$1.62x10^{-5}$
Idaho National Engineering Laboratory	$1.46x10^{-4}$	$1.33x10^{-5}$
Nevada Test Site	$1.83x10^{-4}$	$1.52x10^{-5}$
Savannah River	$7.03x10^{-5}$	$5.08x10^{-6}$
Sweetgrass, MT	$1.67x10^{-4}$	$2.95x10^{-5}$
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	$1.77x10^{-4}$	$6.60x10^{-5}$
Hanford Site	$3.35x10^{-4}$	$7.04x10^{-5}$
Idaho National Engineering Laboratory	$3.16x10^{-4}$	$6.19x10^{-5}$
Nevada Test Site	$3.81x10^{-4}$	$6.60x10^{-5}$
Savannah River	$1.27x10^{-4}$	$1.31x10^{-5}$
Sweetgrass, MT	$3.76x10^{-4}$	$1.14x10^{-4}$
Calcined Target Material		
Alexandria Bay, NY	$1.04x10^{-2}$	$2.20x10^{-3}$
Hanford Site	$3.55x10^{-2}$	$6.78x10^{-3}$
Idaho National Engineering Laboratory	$3.38x10^{-2}$	$6.33x10^{-3}$
Nevada Test Site	$3.50x10^{-2}$	$6.35x10^{-3}$
Savannah River	$5.03x10^{-3}$	$5.03x10^{-4}$
Sweetgrass, MT	$4.51x10^{-2}$	$8.02x10^{-3}$

Shipments to Oak Ridge Reservation:		
Source/Route	***Truck***	***Rail***
Oxidized Target Material		
Alexandria Bay, NY	$2.61x10^{-2}$	$5.51x10^{-3}$
Hanford Site	$8.88x10^{-2}$	$1.70x10^{-2}$
Idaho National Engineering Laboratory	$8.47x10^{-2}$	$1.58x10^{-2}$
Nevada Test Site	$8.75x10^{-2}$	$1.59x10^{-2}$
Savannah River	$1.26x10^{-2}$	$1.26x10^{-3}$
Sweetgrass, MT	$1.13x10^{-1}$	$2.01x10^{-2}$

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
	From Eastern Ports	
BR-2 Belgium Spent Nuclear Fuel		
Charleston, SC (NWS)	$8.42x10^{-6}$	$8.97x10^{-7}$
Charleston, SC (Wando Terminal)	$1.14x10^{-5}$	$8.97x10^{-7}$
Elizabeth, NJ	$5.89x10^{-5}$	$2.50x10^{-5}$
Galveston, TX	$8.20x10^{-5}$	$2.19x10^{-5}$
Jacksonville, FL	$9.46x10^{-6}$	$1.65x10^{-6}$
Newport News, VA	$3.16x10^{-5}$	$5.04x10^{-6}$
Norfolk, VA	$2.55x10^{-5}$	$3.77x10^{-6}$
Philadelphia, PA	$6.56x10^{-5}$	$1.65x10^{-5}$
Portsmouth, VA	$2.59x10^{-5}$	$3.27x10^{-6}$
Savannah, GA	$7.13x10^{-6}$	$5.84x10^{-7}$
MOTSU, NC	$6.37x10^{-6}$	$2.43x10^{-6}$
Wilmington, NC	$1.36x10^{-5}$	$2.32x10^{-6}$
RHF France Spent Nuclear Fuel		
Charleston, SC (NWS)	$3.64x10^{-6}$	$3.87x10^{-7}$
Charleston, SC (Wando Terminal)	$4.95x10^{-6}$	$3.87x10^{-7}$
Elizabeth, NJ	$2.54x10^{-5}$	$1.08x10^{-5}$
Galveston, TX	$3.55x10^{-5}$	$9.46x10^{-6}$
Jacksonville, FL	$4.09x10^{-6}$	$7.13x10^{-7}$
Newport News, VA	$1.37x10^{-5}$	$2.17x10^{-6}$
Norfolk, VA	$1.10x10^{-5}$	$1.63x10^{-6}$
Philadelphia, PA	$2.83x10^{-5}$	$7.13x10^{-6}$
Portsmouth, VA	$1.12x10^{-5}$	$1.41x10^{-6}$
Savannah GA	$3.08x10^{-6}$	$2.52x10^{-7}$
MOTSU NC	$2.75x10^{-6}$	$1.05x10^{-6}$
Wilmington NC	$5.90x10^{-6}$	$1.00x10^{-6}$
NRU Canada Spent Nuclear Fuel		
Charleston, SC (NWS)	$1.16x10^{-5}$	$1.24x10^{-6}$
Charleston, SC (Wando Terminal)	$1.58x10^{-5}$	$1.24x10^{-6}$
Elizabeth, NJ	$8.12x10^{-5}$	$3.45x10^{-5}$
Galveston, TX	$1.13x10^{-4}$	$3.02x10^{-5}$
Jacksonville, FL	$1.31x10^{-5}$	$2.28x10^{-6}$
Newport News, VA	$4.36x10^{-5}$	$6.96x10^{-6}$
Norfolk, VA	$3.52x10^{-5}$	$5.21x10^{-6}$
Philadelphia, PA	$9.04x10^{-5}$	$2.28x10^{-5}$
Portsmouth, VA	$3.57x10^{-5}$	$4.52x10^{-6}$
Savannah, GA	$9.83x10^{-6}$	$8.07x10^{-7}$
MOTSU, NC	$8.79x10^{-6}$	$3.36x10^{-6}$
Wilmington, NC	$1.88x10^{-5}$	$3.21x10^{-6}$
PRR-1 TRIGA Spent Nuclear Fuel		
Charleston, SC (NWS)	$2.14x10^{-5}$	$3.19x10^{-6}$

Shipments to Savannah River Site:		
Source/Route	**Truck**	**Rail**
Charleston, SC (Wando Terminal)	$2.89x10^{-5}$	$3.19x10^{-6}$
Elizabeth, NJ	$1.54x10^{-4}$	$8.32x10^{-5}$
Galveston, TX	$2.05x10^{-4}$	$7.50x10^{-5}$
Jacksonville, FL	$2.56x10^{-5}$	$6.32x10^{-6}$
Newport News, VA	$8.21x10^{-5}$	$1.79x10^{-5}$
Norfolk, VA	$6.71x10^{-5}$	$1.37x10^{-5}$
Philadelphia, PA	$1.70x10^{-4}$	$5.58x10^{-5}$
Portsmouth, VA	$6.80x10^{-5}$	$1.20x10^{-5}$
Savannah, GA	$1.87x10^{-5}$	$2.22x10^{-6}$
MOTSU, NC	$1.73x10^{-5}$	$8.90x10^{-6}$
Wilmington, NC	$3.58x10^{-5}$	$8.52x10^{-6}$
HLW Vitrified		
Charleston, SC (NWS)	$1.27x10^{-4}$	$1.65x10^{-5}$
Charleston, SC (Wando Terminal)	$1.71x10^{-4}$	$1.65x10^{-5}$
Elizabeth, NJ	$1.19x10^{-3}$	$3.16x10^{-4}$
Galveston, TX	$1.21x10^{-3}$	$3.45x10^{-4}$
Jacksonville, FL	$2.93x10^{-4}$	$4.13x10^{-5}$
Newport News, VA	$6.64x10^{-4}$	$9.55x10^{-5}$
Norfolk, VA	$6.23x10^{-4}$	$7.95x10^{-5}$
Philadelphia, PA	$1.25x10^{-3}$	$2.28x10^{-4}$
Portsmouth, VA	$6.25x10^{-4}$	$7.24x10^{-5}$
Savannah, GA	$1.84x10^{-4}$	$1.44x10^{-5}$
MOTSU, NC	$1.98x10^{-4}$	$5.24x10^{-5}$
Wilmington, NC	$3.43x10^{-4}$	$5.04x10^{-5}$
Calcined Target Material		
Charleston, SC (NWS)	$1.15x10^{-3}$	$1.16x 10^{-4}$
Charleston, SC (Wando Terminal)	$1.33x10^{-3}$	$1.16x10^{-4}$
Elizabeth, NJ	$8.28x10^{-3}$	$2.17x10^{-3}$
Jacksonville, FL	$2.09x10^{-3}$	$2.96x10^{-4}$
Newport News, VA	$4.69x10^{-3}$	$6.69x10^{-4}$
Norfolk, VA	$4.41x10^{-3}$	$5.60x10^{-4}$
Philadelphia, PA	$8.79x10^{-3}$	$1.58x10^{-3}$
Portsmouth, VA	$4.42x10^{-3}$	$5.12x10^{-4}$
Savannah, GA	$1.30x10^{-3}$	$1.03x10^{-4}$
MOTSU, NC	$1.42x10^{-3}$	$3.71x10^{-4}$
Wilmington, NC	$2.43x10^{-3}$	$3.57x10^{-4}$
Galveston, TX	$8.43x10^{-3}$	$2.40x10^{-3}$
Oxidized Target Material		
Charleston, SC (NWS)	$2.88x10^{-3}$	$2.90x10^{-4}$
Charleston, SC (Wando Terminal)	$3.33x10^{-3}$	$2.90x10^{-4}$
Elizabeth, NJ	$2.07x10^{-2}$	$5.44x10^{-3}$
Jacksonville, FL	$5.23x10^{-3}$	$7.40x10^{-4}$
Newport News, VA	$1.17x10^{-2}$	$1.68x10^{-3}$
Norfolk, VA	$1.10x10^{-2}$	$1.40x10^{-3}$
Philadelphia, PA	$2.20x10^{-2}$	$3.95x10^{-3}$
Portsmouth, VA	$1.11x10^{-2}$	$1.28x10^{-3}$
Savannah, GA	$3.25x10^{-3}$	$2.57x10^{-4}$
MOTSU, NC	$3.54x10^{-3}$	$9.28x10^{-4}$
Wilmington, NC	$6.08x10^{-3}$	$8.93x10^{-4}$
Galveston, TX	$2.11x10^{-2}$	$6.00x10^{-3}$

Shipments to Savannah River Site:		
Source/Route	***Truck***	***Rail***
	From Western Ports	
BR-2 Belgium Spent Nuclear Fuel		
NWS Concord, CA	1.52×10^{-4}	3.67×10^{-5}
Long Beach, CA	1.37×10^{-4}	3.15×10^{-5}
Portland, OR	1.80×10^{-4}	3.14×10^{-5}
Tacoma, WA	1.68×10^{-4}	3.29×10^{-5}
RHF France Spent Nuclear Fuel		
NWS Concord, CA	6.56×10^{-5}	1.58×10^{-5}
Long Beach, CA	5.94×10^{-5}	1.36×10^{-5}
Portland, OR	7.76×10^{-5}	1.35×10^{-5}
Tacoma, WA	7.24×10^{-5}	1.42×10^{-5}
NRU Canada Spent Nuclear Fuel		
NWS Concord, CA	2.09×10^{-4}	5.07×10^{-5}
Long Beach, CA	1.89×10^{-4}	4.37×10^{-5}
Portland, OR	2.48×10^{-4}	4.37×10^{-5}
Tacoma, WA	2.31×10^{-4}	4.57×10^{-5}
PRR-1 TRIGA Spent Nuclear Fuel		
NWS Concord, CA	3.90×10^{-4}	1.31×10^{-4}
Long Beach, CA	3.51×10^{-4}	1.46×10^{-4}
Portland, OR	5.00×10^{-4}	1.55×10^{-4}
Tacoma, WA	4.96×10^{-4}	1.59×10^{-4}
Calcined Target Material		
NWS Concord, CA	1.81×10^{-2}	4.78×10^{-3}
Long Beach, CA	1.61×10^{-2}	1.01×10^{-2}
Portland, OR	4.05×10^{-2}	1.02×10^{-2}
Tacoma, WA	5.15×10^{-2}	1.03×10^{-2}
Oxidized Target Material		
NWS Concord, CA	4.54×10^{-2}	1.20×10^{-2}
Long Beach, CA	4.03×10^{-2}	2.53×10^{-2}
Portland, OR	1.01×10^{-1}	2.54×10^{-2}
Tacoma, WA	1.29×10^{-1}	2.57×10^{-2}
	From DOE Sites/Canadian Border	
BR-2 Belgium Spent Nuclear Fuel		
Alexandria Bay, NY	6.60×10^{-5}	4.27×10^{-5}
Hanford Site	1.61×10^{-4}	1.97×10^{-5}
Idaho National Engineering Laboratory	1.54×10^{-4}	1.76×10^{-5}
Nevada Test Site	1.81×10^{-4}	1.90×10^{-5}
Oak Ridge Reservation	5.10×10^{-5}	3.67×10^{-6}
Sweetgrass, MT	1.67×10^{-4}	2.85×10^{-5}
RHF France Spent Nuclear Fuel		
Alexandria Bay, NY	2.85×10^{-5}	1.85×10^{-5}
Hanford Site	6.94×10^{-5}	8.47×10^{-6}
Idaho National Engineering Laboratory	6.66×10^{-5}	7.57×10^{-6}
Nevada Test Site	7.81×10^{-5}	8.18×10^{-6}
Oak Ridge Reservation	2.21×10^{-5}	1.59×10^{-6}
Sweetgrass, MT	7.23×10^{-5}	1.23×10^{-5}
NRU Canada Spent Nuclear Fuel		
Alexandria Bay, NY	9.11×10^{-5}	5.90×10^{-5}
Hanford Site	2.22×10^{-4}	2.73×10^{-5}
Idaho National Engineering Laboratory	2.13×10^{-4}	2.44×10^{-5}
Nevada Test Site	2.49×10^{-4}	2.64×10^{-5}

Shipments to Savannah River Site:		
Source/Route	Truck	Rail
Oak Ridge Reservation	7.03×10^{-5}	5.08×10^{-6}
Sweetgrass, MT	2.31×10^{-4}	3.97×10^{-5}
PRR-1 TRIGA Spent Nuclear Fuel		
Alexandria Bay, NY	1.78×10^{-4}	1.43×10^{-4}
Hanford Site	4.53×10^{-4}	9.49×10^{-5}
Idaho National Engineering Laboratory	4.34×10^{-4}	8.63×10^{-5}
Nevada Test Site	4.99×10^{-4}	9.04×10^{-5}
Oak Ridge Reservation	1.27×10^{-4}	1.31×10^{-5}
Sweetgrass, MT	4.90×10^{-4}	1.44×10^{-4}
Calcined Target Material		
Alexandria Bay, NY	1.07×10^{-2}	3.61×10^{-3}
Hanford Site	3.96×10^{-2}	7.11×10^{-3}
Idaho National Engineering Laboratory	3.79×10^{-2}	6.66×10^{-3}
Nevada Test Site	3.91×10^{-2}	6.68×10^{-3}
Oak Ridge Reservation	5.03×10^{-3}	5.03×10^{-4}
Sweetgrass, MT	4.91×10^{-2}	9.69×10^{-3}
Oxidized Target Material		
Alexandria Bay, NY	2.68×10^{-2}	9.05×10^{-3}
Hanford Site	9.90×10^{-2}	1.78×10^{-2}
Idaho National Engineering Laboratory	9.49×10^{-2}	1.67×10^{-2}
Nevada Test Site	9.77×10^{-2}	1.67×10^{-2}
Oak Ridge Reservation	1.26×10^{-2}	1.26×10^{-3}
Sweetgrass, MT	1.23×10^{-1}	2.42×10^{-2}

The nonradiological risk factors are presented in terms of mortalities per shipment in Table E-10. Separate risk factors are provided for mortalities resulting from hydrocarbon emissions and transportation accidents (fatalities resulting from mechanical impact).

Table E-10 Vehicle-Related (Nonradiological) Risk Factors per Shipment to Spent Nuclear Fuel Types (Fatalities/Shipment)

Shipments to Hanford Site:			
Mode		Emission	Accident
From Eastern Ports			
Truck			
	Charleston, SC (NWS)	1.11×10^{-5}	2.00×10^{-4}
	Charleston, SC (Wando Terminal)	1.18×10^{-5}	2.01×10^{-4}
	Elizabeth, NJ	1.31×10^{-5}	1.66×10^{-4}
	Galveston, TX	1.69×10^{-5}	1.50×10^{-4}
	Jacksonville, FL	1.44×10^{-5}	1.95×10^{-4}
	Newport News, VA	1.66×10^{-5}	1.80×10^{-4}
	Norfolk, VA	1.64×10^{-5}	1.83×10^{-4}
	Philadelphia, PA	1.42×10^{-5}	1.65×10^{-4}
	Portsmouth, VA	1.82×10^{-5}	1.82×10^{-4}
	Savannah, GA	1.36×10^{-5}	1.86×10^{-4}
	Sunny Point, NC	1.17×10^{-5}	1.82×10^{-4}
	Wilmington, NC	1.09×10^{-5}	2.10×10^{-4}
Rail			
	Charleston, SC (NWS)	2.35×10^{-5}	6.40×10^{-6}
	Charleston, SC (Wando Terminal)	2.35×10^{-5}	6.40×10^{-6}
	Elizabeth, NJ	5.58×10^{-5}	6.30×10^{-6}
	Galveston, TX	9.83×10^{-6}	5.00×10^{-6}

Shipments to Hanford Site:			
Mode		***Emission***	***Accident***
	Jacksonville, FL	2.01×10^{-5}	6.42×10^{-6}
	Newport News, VA	3.49×10^{-5}	6.46×10^{-6}
	Norfolk, VA	3.56×10^{-5}	6.67×10^{-6}
	Philadelphia, PA	5.38×10^{-5}	6.20×10^{-6}
	Portsmouth, VA	3.46×10^{-5}	6.60×10^{-6}
	Savannah, GA	1.86×10^{-5}	6.47×10^{-6}
	Sunny Point, NC	2.07×10^{-5}	6.70×10^{-6}
	Wilmington, NC	2.07×10^{-5}	6.68×10^{-6}
From Western Ports			
Truck			
	Concord, CA	7.18×10^{-6}	4.81×10^{-5}
	Long Beach, CA	2.08×10^{-5}	7.74×10^{-5}
	Portland, OR	2.67×10^{-6}	1.04×10^{-5}
	Tacoma, WA	3.06×10^{-6}	1.03×10^{-5}
Rail			
	Concord, CA	1.99×10^{-5}	1.99×10^{-6}
	Long Beach, CA	3.67×10^{-5}	3.32×10^{-6}
	Portland, OR	4.48×10^{-6}	5.00×10^{-7}
	Tacoma, WA	5.61×10^{-6}	7.82×10^{-7}
From DOE Sites/Canadian Border			
Truck			
	Alexandria Bay, NY	1.46×10^{-5}	1.49×10^{-4}
	Idaho National Engineering Laboratory	2.19×10^{-6}	3.07×10^{-5}
	Nevada Test Site	9.50×10^{-6}	6.38×10^{-5}
	Oak Ridge Reservation	9.50×10^{-6}	1.61×10^{-4}
	Savannah River	1.31×10^{-5}	1.81×10^{-4}
	Sweetgrass, MT	1.74×10^{-6}	4.15×10^{-5}
Rail			
	Alexandria Bay, NY	4.59×10^{-5}	6.02×10^{-6}
	Idaho National Engineering Laboratory	3.98×10^{-6}	1.38×10^{-6}
	Nevada Test Site	5.90×10^{-6}	2.72×10^{-6}
	Oak Ridge Reservation	1.45×10^{-5}	5.44×10^{-6}
	Savannah River	2.21×10^{-5}	6.18×10^{-6}
	Sweetgrass MT	3.98×10^{-6}	1.27×10^{-6}

Shipments to Idaho National Engineering Laboratory:			
Mode		***Emission***	***Accident***
From Eastern Ports			
Truck			
	Charleston, SC (NWS)	1.03×10^{-5}	1.80×10^{-4}
	Charleston, SC (Wando Terminal)	1.09×10^{-5}	1.81×10^{-4}
	Elizabeth, NJ	1.17×10^{-5}	1.46×10^{-4}
	Galveston, TX	1.54×10^{-5}	1.30×10^{-4}
	Jacksonville, FL	1.22×10^{-5}	1.75×10^{-4}
	Newport News, VA	1.52×10^{-5}	1.60×10^{-4}
	Norfolk, VA	1.43×10^{-5}	1.63×10^{-4}
	Philadelphia, PA	1.28×10^{-5}	1.45×10^{-4}
	Portsmouth, VA	1.67×10^{-5}	1.62×10^{-4}
	Savannah, GA	1.22×10^{-5}	1.66×10^{-4}
	Sunny Point, NC	9.53×10^{-6}	1.59×10^{-4}
	Wilmington, NC	9.50×10^{-6}	1.89×10^{-4}

Shipments to Idaho National Engineering Laboratory:			
Mode		**Emission**	**Accident**
Rail			
	Charleston, SC (NWS)	2.16×10^{-5}	5.26×10^{-6}
	Charleston, SC (Wando Terminal)	2.16×10^{-5}	5.26×10^{-6}
	Elizabeth, NJ	5.39×10^{-5}	5.15×10^{-6}
	Galveston, TX	7.91×10^{-6}	3.86×10^{-6}
	Jacksonville, FL	1.82×10^{-5}	5.28×10^{-6}
	Newport News, VA	3.30×10^{-5}	5.32×10^{-6}
	Norfolk, VA	3.37×10^{-5}	5.53×10^{-6}
	Philadelphia, PA	5.19×10^{-5}	5.05×10^{-6}
	Portsmouth, VA	3.26×10^{-5}	5.46×10^{-6}
	Savannah, GA	1.67×10^{-5}	5.33×10^{-6}
	Sunny Point, NC	1.88×10^{-5}	5.56×10^{-6}
	Wilmington, NC	1.88×10^{-5}	5.54×10^{-6}
From Western Ports			
Truck			
	Concord, CA	9.40×10^{-6}	5.52×10^{-5}
	Long Beach, CA	2.55×10^{-5}	6.21×10^{-5}
	Portland, OR	3.93×10^{-6}	3.62×10^{-5}
	Sweetgrass, MT	7.08×10^{-7}	2.89×10^{-5}
	Tacoma, WA	4.28×10^{-6}	3.97×10^{-5}
Rail			
	Concord, CA	9.08×10^{-6}	1.91×10^{-6}
	Long Beach, CA	3.48×10^{-5}	2.18×10^{-6}
	Portland, OR	5.36×10^{-6}	1.64×10^{-6}
	Sweetgrass, MT	5.06×10^{-6}	2.58×10^{-6}
	Tacoma, WA	8.62×10^{-6}	1.96×10^{-6}
From DOE Sites/Canadian Border			
Truck			
	Alexandria Bay, NY	1.32×10^{-5}	1.29×10^{-4}
	Hanford Site	2.19×10^{-6}	3.07×10^{-5}
	Nevada Test Site	8.08×10^{-6}	4.36×10^{-5}
	Oak Ridge Reservation	8.08×10^{-6}	1.41×10^{-4}
	Savannah River	1.17×10^{-5}	1.61×10^{-4}
Rail			
	Alexandria Bay, NY	4.40×10^{-5}	4.88×10^{-6}
	Hanford Site	3.98×10^{-6}	1.38×10^{-6}
	Nevada Test Site	3.98×10^{-6}	1.58×10^{-6}
	Oak Ridge Reservation	1.26×10^{-5}	4.30×10^{-6}
	Savannah River	2.01×10^{-5}	5.04×10^{-6}

Shipments to Nevada Test Site:			
Mode		**Emission**	**Accident**
From Eastern Ports			
Truck			
	Charleston, SC (NWS)	1.16×10^{-5}	1.92×10^{-4}
	Charleston, SC (Wando Terminal)	1.23×10^{-5}	1.94×10^{-4}
	Elizabeth, NJ	1.98×10^{-5}	1.87×10^{-4}
	Galveston, TX	1.90×10^{-5}	1.32×10^{-4}
	Jacksonville, FL	1.49×10^{-5}	1.88×10^{-4}
	Newport News, VA	1.79×10^{-5}	1.74×10^{-4}
	Norfolk, VA	1.70×10^{-5}	1.76×10^{-4}
	Philadelphia, PA	1.90×10^{-5}	1.85×10^{-4}

Shipments to Nevada Test Site:			
Mode		***Emission***	***Accident***
	Portsmouth, VA	$1.94x10^{-5}$	$1.75x10^{-4}$
	Savannah, GA	$1.49x10^{-5}$	$1.79x10^{-4}$
	Sunny Point, NC	$1.57x10^{-5}$	$1.90x10^{-4}$
	Wilmington, NC	$1.22x10^{-5}$	$2.03x10^{-4}$
Rail			
	Charleston, SC (NWS)	$2.42x10^{-5}$	$6.16x10^{-6}$
	Charleston, SC (Wando Terminal)	$2.42x10^{-5}$	$6.16x10^{-6}$
	Elizabeth, NJ	$5.64x10^{-5}$	$6.06x10^{-6}$
	Galveston, TX	$6.44x10^{-6}$	$4.09x10^{-6}$
	Jacksonville, FL	$2.08x10^{-5}$	$6.18x10^{-6}$
	Newport News, VA	$3.56x10^{-5}$	$6.22x10^{-6}$
	Norfolk, VA	$3.63x10^{-5}$	$6.43x10^{-6}$
	Philadelphia, PA	$5.45x10^{-5}$	$5.96x10^{-6}$
	Portsmouth, VA	$3.52x10^{-5}$	$6.37x10^{-6}$
	Savannah, GA	$1.92x10^{-5}$	$6.23x10^{-6}$
	Sunny Point, NC	$2.14x10^{-5}$	$6.46x10^{-6}$
	Wilmington, NC	$2.14x10^{-5}$	$6.44x10^{-6}$
From Western Ports			
Truck			
	Concord, CA	$1.59x10^{-5}$	$5.34x10^{-5}$
	Long Beach, CA	$2.06x10^{-5}$	$2.83x10^{-5}$
	Portland, OR	$1.20x10^{-5}$	$6.95x10^{-5}$
	Tacoma, WA	$1.16x10^{-5}$	$7.28x10^{-5}$
Rail			
	Concord, CA	$1.97x10^{-5}$	$1.78x10^{-6}$
	Long Beach, CA	$3.08x10^{-5}$	$1.01x10^{-6}$
	Portland, OR	$7.28x10^{-6}$	$2.99x10^{-6}$
	Tacoma, WA	$1.05x10^{-5}$	$3.30x10^{-6}$
From DOE Sites/Canadian Border			
Truck			
	Alexandria Bay, NY	$1.62x10^{-5}$	$1.58x10^{-4}$
	Hanford Site	$9.50x10^{-6}$	$6.38x10^{-5}$
	Idaho National Engineering Laboratory	$8.08x10^{-6}$	$4.36x10^{-5}$
	Oak Ridge Reservation	$1.08x10^{-5}$	$1.54x10^{-4}$
	Savannah River	$1.43x10^{-5}$	$1.74x10^{-4}$
	Sweetgrass, MT	$9.59x10^{-6}$	$6.77x10^{-5}$
Rail			
	Alexandria Bay, NY	$4.66x10^{-5}$	$5.78x10^{-6}$
	Hanford Site	$5.90x10^{-6}$	$2.72x10^{-6}$
	Idaho National Engineering Laboratory	$3.98x10^{-6}$	$1.58x10^{-6}$
	Oak Ridge Reservation	$1.52x10^{-5}$	$5.20x10^{-6}$
	Savannah River	$2.27x10^{-5}$	$5.94x10^{-6}$
	Sweetgrass, MT	$7.03x10^{-6}$	$3.92x10^{-6}$

Shipments to Oak Ridge Reservation:			
Mode		***Emission***	***Accident***
From Eastern Ports			
Truck			
	Charleston, SC (NWS)	$1.06x10^{-6}$	$3.81x10^{-5}$
	Charleston, SC (Wando Terminal)	$1.74x10^{-6}$	$3.94x10^{-5}$
	Elizabeth, NJ	$4.89x10^{-6}$	$5.42x10^{-5}$
	Galveston, TX	$6.60x10^{-6}$	$7.19x10^{-5}$

Shipments to Oak Ridge Reservation:			
Mode		Emission	Accident
	Jacksonville, FL	2.29×10^{-6}	4.23×10^{-5}
	Newport News, VA	4.67×10^{-6}	3.66×10^{-5}
	Norfolk, VA	2.32×10^{-6}	3.64×10^{-5}
	Philadelphia, PA	7.95×10^{-6}	4.81×10^{-5}
	Portsmouth, VA	6.21×10^{-6}	3.79×10^{-5}
	Savannah, GA	9.33×10^{-7}	4.18×10^{-5}
	Sunny Point, NC	1.38×10^{-6}	4.66×10^{-5}
	Wilmington, NC	1.42×10^{-6}	4.94×10^{-5}
Rail			
	Charleston, SC (NWS)	3.60×10^{-6}	1.22×10^{-6}
	Charleston, SC (Wando Terminal)	3.60×10^{-6}	1.22×10^{-6}
	Elizabeth, NJ	4.00×10^{-5}	1.64×10^{-6}
	Galveston, TX	1.46×10^{-5}	2.20×10^{-6}
	Jacksonville, FL	5.61×10^{-6}	1.18×10^{-6}
	Newport News, VA	6.44×10^{-6}	1.60×10^{-6}
	Norfolk, VA	4.48×10^{-6}	1.44×10^{-6}
	Philadelphia, PA	2.46×10^{-5}	1.47×10^{-6}
	Portsmouth, VA	3.47×10^{-6}	1.38×10^{-6}
	Savannah, GA	4.06×10^{-6}	1.23×10^{-6}
	Sunny Point, NC	3.35×10^{-6}	1.13×10^{-6}
	Wilmington, NC	3.35×10^{-6}	1.11×10^{-6}
From Western Ports			
Truck			
	Concord, CA	2.31×10^{-5}	1.95×10^{-4}
	Long Beach, CA	2.78×10^{-5}	1.70×10^{-4}
	Portland, OR	1.25×10^{-5}	1.68×10^{-4}
	Tacoma, WA	8.56×10^{-6}	1.38×10^{-4}
Rail			
	Concord, CA	2.53×10^{-5}	5.88×10^{-6}
	Long Beach, CA	4.33×10^{-5}	5.59×10^{-6}
	Portland, OR	2.87×10^{-5}	5.91×10^{-6}
	Tacoma, WA	3.56×10^{-5}	5.93×10^{-6}
From DOE Sites/Canadian Border			
Truck			
	Alexandria Bay, NY	1.96×10^{-6}	6.31×10^{-5}
	Hanford Site	9.50×10^{-6}	1.61×10^{-4}
	Idaho National Engineering Laboratory	8.08×10^{-6}	1.41×10^{-4}
	Nevada Test Site	1.08×10^{-5}	1.54×10^{-4}
	Savannah River	2.96×10^{-6}	2.92×10^{-5}
	Sweetgrass, MT	6.98×10^{-6}	1.24×10^{-4}
Rail			
	Alexandria Bay, NY	2.78×10^{-5}	2.03×10^{-6}
	Hanford Site	1.45×10^{-5}	5.44×10^{-6}
	Idaho National Engineering Laboratory	1.26×10^{-5}	4.30×10^{-6}
	Nevada Test Site	1.52×10^{-5}	5.20×10^{-6}
	Savannah River	2.51×10^{-6}	8.72×10^{-7}
	Sweetgrass, MT	2.31×10^{-5}	4.39×10^{-6}

Shipments to Savannah River Site:			
Mode		***Emission***	***Accident***
	From Eastern Ports		
Truck			
	Charleston, SC (NWS)	5.15×10^{-7}	1.61×10^{-5}
	Charleston, SC (Wando Terminal)	1.19×10^{-6}	1.74×10^{-5}
	Elizabeth, NJ	5.47×10^{-6}	6.58×10^{-5}
	Galveston, TX	8.11×10^{-6}	7.15×10^{-5}
	Jacksonville, FL	3.22×10^{-8}	2.94×10^{-5}
	Newport News, VA	3.54×10^{-6}	4.57×10^{-5}
	Norfolk, VA	1.58×10^{-6}	4.45×10^{-5}
	Philadelphia, PA	9.37×10^{-6}	6.30×10^{-5}
	Portsmouth, VA	1.83×10^{-6}	4.47×10^{-5}
	Savannah, GA	3.22×10^{-8}	2.10×10^{-5}
	Sunny Point, NC	2.57×10^{-7}	2.17×10^{-5}
	Wilmington, NC	5.15×10^{-7}	2.87×10^{-5}
Rail			
	Charleston, SC (NWS)	1.46×10^{-6}	2.93×10^{-7}
	Charleston, SC (Wando Terminal)	1.46×10^{-6}	2.93×10^{-7}
	Elizabeth, NJ	3.92×10^{-5}	1.82×10^{-6}
	Galveston, TX	2.06×10^{-5}	2.46×10^{-6}
	Jacksonville, FL	2.80×10^{-6}	5.42×10^{-7}
	Newport News, VA	5.61×10^{-6}	1.26×10^{-6}
	Norfolk, VA	3.64×10^{-6}	1.11×10^{-6}
	Philadelphia, PA	2.39×10^{-5}	1.65×10^{-6}
	Portsmouth, VA	2.64×10^{-6}	1.04×10^{-6}
	Savannah, GA	5.86×10^{-7}	2.38×10^{-7}
	Sunny Point, NC	2.55×10^{-6}	7.99×10^{-7}
	Wilmington, NC	2.55×10^{-6}	7.80×10^{-7}
	From Western Ports		
Truck			
	Concord, CA	2.99×10^{-5}	1.96×10^{-4}
	Long Beach, CA	2.57×10^{-5}	1.68×10^{-4}
	Portland, OR	1.60×10^{-5}	1.88×10^{-4}
	Tacoma, WA	1.21×10^{-5}	1.59×10^{-4}
Rail			
	Concord, CA	4.80×10^{-5}	6.66×10^{-6}
	Long Beach, CA	5.07×10^{-5}	6.78×10^{-6}
	Portland, OR	3.44×10^{-5}	6.60×10^{-6}
	Tacoma, WA	4.13×10^{-5}	6.62×10^{-6}
	From DOE Sites/Canadian Border		
Truck			
	Alexandria Bay, NY	2.54×10^{-6}	7.47×10^{-5}
	Hanford Site	1.31×10^{-5}	1.81×10^{-4}
	Idaho National Engineering Laboratory	1.17×10^{-5}	1.61×10^{-4}
	Nevada Test Site	1.43×10^{-5}	1.74×10^{-4}
	Oak Ridge Reservation	2.96×10^{-6}	2.92×10^{-5}
	Sweetgrass, MT	1.05×10^{-5}	1.43×10^{-4}
Rail			
	Alexandria Bay, NY	5.76×10^{-5}	2.68×10^{-6}
	Hanford Site	2.21×10^{-5}	6.18×10^{-6}
	Idaho National Engineering Laboratory	2.01×10^{-5}	5.04×10^{-6}
	Nevada Test Site	2.27×10^{-5}	5.94×10^{-6}

Shipments to Savannah River Site:			
	Mode	Emission	Accident
	Oak Ridge Reservation	2.51×10^{-6}	8.72×10^{-7}
	Sweetgrass, MT	2.87×10^{-5}	5.07×10^{-6}

The total risks for any alternative or option can be calculated by multiplying the number of foreign research reactor spent nuclear fuel shipments by the per-shipment risk factors provided in Tables E-8 through E-10.

E.7.1.2 Characterization of Shipment Risks

The results of the per-shipment analysis are shown in Tables E-8 through E-10. From these tables, it is clear that the incident-free dose would be much higher than the accident dose for each of the fuel types. The accident doses are based on realistic, yet conservative fuel loadings. Since most of the public dose would be from incident-free exposure, it is not overly conservative to assume, for assessment purposes, that all spent nuclear fuel can be represented by the fuel type with the highest risk factors for the remainder of the transportation analysis.

E.7.2 Evaluation of the Basic Implementation

The following sections describe the evaluation of the basic implementation of the Management Alternative 1 of the proposed action. The evaluation of the management and implementation alternatives are described in Section E.8.

E.7.2.1 Shipments

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of 837 casks from points of entry (marine ports and Canadian border crossings) to DOE facilities. The basic assumption used in determining the number of shipments is that spent nuclear fuel from countries bordering the Atlantic Ocean and Mediterranean Sea was assumed to arrive on the east coast of the United States, and spent nuclear fuel from countries bordering the Indian and Pacific Oceans was assumed to arrive on the west coast. This is conservative from an overland transportation standpoint, because, as shown in Tables E-8 through E-10, shipment to the coast nearest the management site would reduce the risk factors for the overland shipment. Additionally, this assumption is considered to be realistic because the long shipping times required to ship from the Pacific Ocean to east coast ports and from the Atlantic Ocean to west coast ports, would be costly in terms of shipping, and would tie up the world's already short supply of casks. The foreign research reactor spent nuclear fuel could arrive at any port that meets the criteria identified in Appendix D, and would be likely to arrive at a variety of these ports. The basic shipment count, by point of origin is:

	East Coast		West Coast		
	Aluminum	TRIGA	Aluminum	TRIGA	Totals
Phase 1	419	82	101	42	644
Phase 2	125	25	30	13	193
Totals	544	107	131	55	837

Several of the SNF&INEL Final EIS (DOE, 1995) alternatives involve consolidation of all spent nuclear fuel to Idaho National Engineering Laboratory and/or Savannah River Site and, therefore, are single-phase programs that would require no additional shipments. However, many of the possible options require the use of Hanford Site, Nevada Test Site and/or Oak Ridge Reservation; and, thus, would require intersite shipments. The number of intersite shipments is calculated based on the assumption that the equivalent of

10 seagoing foreign research reactor casks would fit into a single rail cask that would travel between DOE sites. Similarly, it is assumed that the contents of four foreign research reactor casks would fit into a single truck cask for intersite shipment. This is based on the distribution of cask capacities described in Appendix B. As described in Appendix B, there is considerable uncertainty in what storage mode would be used at the Phase 1 site, and therefore in what form the fuel would be for intersite shipment. Additionally, it is not clear what casks would be licensed and available when the intersite shipments would begin (approximately 2006). Therefore, these assumptions, which are neither definitely conservative nor nonconservative, are considered to be reasonable and realistic.

The number of intersite shipments for SNF&INEL Final EIS (DOE, 1995) alternatives that would require two-phased approaches varies between 13 and 161. The variation is caused by the large number of unique combinations of Phase 1 and Phase 2 approaches depending on the specific management sites selected. Additionally, the variation is affected by the assumption that larger truck and rail casks would be used for intersite shipments. The actual numbers of shipments are shown in Tables E-1 and E-2.

E.7.2.2 Evaluation Using Risk Factors

Since the fuel would actually arrive at a variety of ports, average shipment risk factors were calculated for east coast ports to each DOE site, and an average shipment risk factor for west coast ports to each DOE site. This approach does not require that a specific port be selected for analysis purposes. It instead models the average affect the foreign research reactor spent nuclear fuel acceptance policy might actually have on the public. This approach is conservative since the dose rates and curie content of the fuel used for the analyses were selected to be conservative, but as realistic as possible, since it is impossible to predict the distribution of shipments among the capable ports.

The upper and lower bound risk estimates for the foreign research reactor spent nuclear fuel policy were also calculated. The upper bound assumes that DOE chooses the acceptable port with the highest per-shipment risk factors for all shipments, and the lower bound risk estimates assume that DOE chooses the acceptable port with the lowest per-shipment risk factors. In general, the highest risk factors result from the longest shipments, and the smallest risk factors from the shortest shipments.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.013 to 0.30 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCF to the public and the crews.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel, 2) combinations of Phase 1 and Phase 2 sites that create varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.006 to 0.071. The shipment by truck would yield higher crew exposures than the shipment by rail since the truck drivers would tend to sit closer to the cask than engineers. Doses to inspectors, security guards, and rail switchyard workers are also considered.

Truck and rail crew members are not radiation workers and, therefore, are not allowed to exceed a dosage of 100 mrem per yr. The regulatory limit for dose rate in occupied areas of the truck or train is 2 mrem per hr. Since a cross-country trip can take just over 50 hr of driving, if the radiation levels were at the maximum allowed, a driver could exceed his or her annual limit. Therefore, DOE would implement administrative controls beyond those required by Federal regulations to ensure that vehicle operators would not exceed their annual dose limits.

The public would be exposed to a small amount of radiation emanating from the cask, and also to pollutants associated with the diesel exhaust. The estimated number of radiation-related LCFs for the general population ranged from 0.007 to 0.22, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.05. The fact that all these risk numbers are less than one means that the basic implementation would be unlikely to increase the total number of individuals that die of cancer in the United States (there are approximately 300,000 cancer deaths per yr in the United States) by a single fatality.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00028 LCF from radiation and from 0.001 to 0.14 for traffic fatality, depending on the transportation mode and DOE sites selected. The reason for the range of fatality estimates is the same as those described for incident-free transportation. These risks, especially in the case of radiological accident risks, are much lower than those for incident-free transportation. The risk estimates are probabilistic, which means that they take the probability of an accident's occurring and the consequences of these accidents into account. The risk estimates indicate that the likelihood of a death or an injury from a vehicle accident not involving radiation or radioactive release would be much higher than a death from a radiation-related accident. Both indicate an expectation of less than one fatality.

The impacts of overland transportation for all alternatives and options are shown in Tables E-11 through E-19. As shown in Tables E-1 and E-2, there are 35 distinct approaches to the basic implementation of Management Alternative 1 of the proposed action. These 35 approaches are all the Phase 1/Phase 2 combinations allowed by the SNF&INEL Final EIS (DOE, 1995). Each of these 35 approaches is evaluated for three different transportation mode assumptions: 1) all shipments are on trucks, 2) that shipments from ports to sites are on trucks and intersite shipments are on rail, and 3) that all shipments are on rail. The transportation mode assumptions of all by truck and all by rail are analyzed to bound the risks of any combination of transportation modes. The third mode assumption is provided as an example of a realistic approach. Each distinct approach and mode assumption is evaluated using the average, upper bound, and lower bound risk factors.

These tables are designed to provide risk estimate factors for all expected implementation alternatives. For example, if the SNF&INEL Final EIS alternative selected is Centralization to Nevada Test Site, "Centralization" should be in the first column of each table, and "Nevada Test Site" in the second column of each table. The Phase 1 approaches available are listed in the third column. The decision as to which of the possible Phase 1 approaches would be used will be part of the foreign research reactor spent nuclear fuel policy described in this EIS. The risk estimates for the foreign research reactor spent nuclear fuel EIS policy are given for "Geographic" distribution of spent nuclear fuel during Phase 1 (to Idaho National Engineering Laboratory and Savannah River Site), for "By Fuel" distribution of spent nuclear fuel during Phase 1 (TRIGA to Idaho National Engineering Laboratory and aluminum-based to Savannah River Site), for "All to Idaho National Engineering Laboratory" during Phase 1, and for "All to Savannah River Site" during Phase 1. The risks, expressed in LCF and traffic accident fatalities are provided. These risk estimates include Phase 1 port-to-site shipments (Savannah River Site and/or Idaho National Engineering Laboratory), intersite shipments to, in this case, Nevada Test Site, and Phase 2 port-to-Nevada Test Site

Table E-11 Tabulation of Overland Transportation Risks: Basic Implementation. All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine			Accidental	
			Radiological		Nonradiological		Radio-logical
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
1992/1993 Planning Basis	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
Regionalization by Fuel Type	INEL/SRS		0.036	0.111	0.005	0.067	0.00004
Regionalization	INEL/SRS		0.019	0.056	0.002	0.035	0.00002
by	INEL/ORR	Geographic	0.022	0.063	0.003	0.040	0.00005
Geography		By Fuel	0.035	0.105	0.005	0.064	0.00007
		All to INEL	0.052	0.165	0.008	0.095	0.00006
	NTS/SRS	Geographic	0.020	0.060	0.003	0.038	0.00003
		By Fuel	0.033	0.102	0.005	0.062	0.00004
		All to SRS	0.030	0.091	0.005	0.056	0.00003
	NTS/ORR	Geographic	0.023	0.068	0.003	0.042	0.00006
		By Fuel	0.036	0.110	0.006	0.067	0.00008
		All to INEL	0.057	0.179	0.009	0.103	0.00009
		All to SRS	0.033	0.100	0.005	0.061	0.00007
	HS/SRS	Geographic	0.019	0.056	0.002	0.035	0.00002
		By Fuel	0.032	0.098	0.005	0.060	0.00004
		All to SRS	0.029	0.087	0.004	0.054	0.00003
	HS/ORR	Geographic	0.022	0.064	0.003	0.040	0.00005
		By Fuel	0.035	0.106	0.005	0.065	0.00007
		All to INEL	0.055	0.173	0.008	0.099	0.00007
		All to SRS	0.032	0.096	0.005	0.059	0.00007
Centralization	INEL		0.062	0.195	0.009	0.112	0.00007
	SRS		0.033	0.097	0.005	0.061	0.00003
	HS	Geographic	0.043	0.134	0.006	0.079	0.00015
		By Fuel	0.057	0.177	0.008	0.104	0.00017
		All to INEL	0.066	0.211	0.010	0.119	0.00008
		All to SRS	0.056	0.176	0.008	0.104	0.00019
	NTS	Geographic	0.042	0.127	0.007	0.080	0.00017
		By Fuel	0.055	0.170	0.009	0.105	0.00019
		All to INEL	0.067	0.212	0.011	0.123	0.00011
		All to SRS	0.054	0.167	0.009	0.104	0.00021
	ORR	Geographic	0.027	0.080	0.003	0.050	0.00008
		By Fuel	0.040	0.121	0.006	0.074	0.00009
		All to INEL	0.066	0.210	0.010	0.123	0.00016
		All to SRS	0.036	0.107	0.005	0.066	0.00008

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-12 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.020	0.058	0.003	0.036	0.00002
		By Fuel	0.033	0.100	0.005	0.061	0.00004
		All to INEL	0.052	0.165	0.008	0.095	0.00006
	NTS/SRS	Geographic	0.019	0.057	0.003	0.036	0.00002
		By Fuel	0.033	0.100	0.005	0.061	0.00003
		All to SRS	0.030	0.091	0.005	0.056	0.00003
	NTS/ORR	Geographic	0.020	0.059	0.003	0.037	0.00002
		By Fuel	0.034	0.102	0.005	0.062	0.00004
		All to INEL	0.054	0.166	0.008	0.096	0.00006
		All to SRS	0.031	0.093	0.005	0.057	0.00004
	HS/SRS	Geographic	0.018	0.054	0.002	0.034	0.00002
		By Fuel	0.032	0.096	0.005	0.059	0.00003
		All to SRS	0.029	0.087	0.004	0.054	0.00003
	HS/ORR	Geographic	0.019	0.056	0.003	0.035	0.00002
		By Fuel	0.033	0.098	0.005	0.060	0.00004
		All to INEL	0.053	0.163	0.008	0.095	0.00006
		All to SRS	0.030	0.089	0.004	0.055	0.00004
Centralization	INEL						
	SRS						
	HS	Geographic	0.032	0.094	0.005	0.055	0.00005
		By Fuel	0.045	0.136	0.008	0.080	0.00007
		All to INEL	0.064	0.200	0.010	0.114	0.00007
		All to SRS	0.042	0.128	0.007	0.075	0.00007
	NTS	Geographic	0.031	0.093	0.006	0.057	0.00006
		By Fuel	0.044	0.135	0.008	0.081	0.00007
		All to INEL	0.063	0.199	0.010	0.116	0.00007
		All to SRS	0.042	0.126	0.008	0.076	0.00008
	ORR	Geographic	0.023	0.067	0.003	0.041	0.00003
		By Fuel	0.036	0.109	0.005	0.066	0.00005
		All to INEL	0.056	0.174	0.009	0.101	0.00008
		All to SRS	0.033	0.100	0.005	0.061	0.00004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-13 Tabulation of Overland Transportation Risks: Basic Implementation. All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
1992/1993 Planning Basis	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
Regionalization by Fuel Type	INEL/SRS		0.011	0.014	0.018	0.002	0.00001
Regionalization by Geography	INEL/SRS		0.008	0.009	0.009	0.001	0.00001
	INEL/ORR	Geographic	0.009	0.013	0.010	0.003	0.00001
		By Fuel	0.012	0.016	0.015	0.004	0.00001
		All to INEL	0.015	0.018	0.019	0.005	0.00001
	NTS/SRS	Geographic	0.009	0.014	0.011	0.004	0.00001
		By Fuel	0.012	0.018	0.016	0.005	0.00001
		All to SRS	0.011	0.018	0.016	0.005	0.00001
	NTS/ORR	Geographic	0.011	0.020	0.011	0.008	0.00004
		By Fuel	0.013	0.020	0.016	0.006	0.00002
		All to INEL	0.020	0.033	0.021	0.013	0.00005
		All to SRS	0.012	0.018	0.015	0.005	0.00002
	HS/SRS	Geographic	0.008	0.011	0.011	0.002	0.00001
		By Fuel	0.011	0.015	0.016	0.003	0.00001
		All to SRS	0.010	0.014	0.015	0.003	0.00001
	HS/ORR	Geographic	0.010	0.016	0.010	0.006	0.00004
		By Fuel	0.012	0.017	0.015	0.004	0.00002
		All to INEL	0.018	0.027	0.019	0.009	0.00002
		All to SRS	0.011	0.014	0.015	0.003	0.00001
Centralization	INEL		0.016	0.016	0.023	0.004	0.00002
	SRS		0.011	0.013	0.017	0.002	0.00001
	HS	Geographic	0.023	0.052	0.015	0.026	0.00012
		By Fuel	0.015	0.020	0.020	0.005	0.00004
		All to INEL	0.020	0.030	0.023	0.010	0.00003
		All to SRS	0.014	0.018	0.020	0.004	0.00004
	NTS	Geographic	0.022	0.049	0.016	0.027	0.00014
		By Fuel	0.016	0.024	0.021	0.007	0.00004
		All to INEL	0.022	0.036	0.024	0.014	0.00005
		All to SRS	0.015	0.021	0.020	0.006	0.00004
	ORR	Geographic	0.014	0.027	0.011	0.012	0.00005
		By Fuel	0.016	0.032	0.016	0.014	0.00004
		All to INEL	0.029	0.064	0.021	0.033	0.00012
		All to SRS	0.014	0.025	0.015	0.009	0.00002

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-14 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Truck, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
1992/1993 Planning Basis	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
Regionalization by Fuel Type	INEL/SRS		0.030	0.093	0.003	0.052	0.000012
Regionalization by Geography	INEL/SRS		0.013	0.040	0.001	0.024	0.000007
	INEL/ORR	Geographic	0.017	0.049	0.002	0.030	0.000039
		By Fuel	0.029	0.089	0.003	0.052	0.000045
		All to INEL	0.044	0.138	0.006	0.078	0.000020
	NTS/SRS	Geographic	0.014	0.043	0.002	0.026	0.000015
		By Fuel	0.027	0.084	0.003	0.048	0.000018
		All to SRS	0.025	0.075	0.003	0.042	0.000011
	NTS/ORR	Geographic	0.018	0.052	0.002	0.032	0.000048
		By Fuel	0.030	0.092	0.004	0.054	0.000052
		All to INEL	0.048	0.151	0.008	0.086	0.000055
		All to SRS	0.028	0.085	0.003	0.049	0.000053
	HS/SRS	Geographic	0.013	0.040	0.001	0.024	0.000009
		By Fuel	0.026	0.081	0.003	0.046	0.000013
		All to SRS	0.024	0.072	0.002	0.040	0.000011
	HS/ORR	Geographic	0.017	0.049	0.002	0.030	0.000041
		By Fuel	0.029	0.089	0.003	0.052	0.000047
		All to INEL	0.046	0.146	0.006	0.082	0.000029
		All to SRS	0.028	0.082	0.003	0.047	0.000052
Centralization	INEL		0.051	0.163	0.007	0.091	0.000023
	SRS		0.028	0.085	0.003	0.047	0.000012
	HS	Geographic	0.036	0.114	0.004	0.065	0.000127
		By Fuel	0.050	0.156	0.006	0.088	0.000135
		All to INEL	0.056	0.179	0.008	0.098	0.000032
		All to SRS	0.050	0.157	0.006	0.088	0.000161
	NTS	Geographic	0.034	0.105	0.005	0.065	0.000145
		By Fuel	0.047	0.146	0.007	0.087	0.000152
		All to INEL	0.056	0.177	0.009	0.100	0.000059
		All to SRS	0.047	0.146	0.007	0.087	0.000176
	ORR	Geographic	0.022	0.066	0.002	0.039	0.000063
		By Fuel	0.034	0.105	0.004	0.060	0.000065
		All to INEL	0.058	0.184	0.007	0.105	0.000122
		All to SRS	0.031	0.094	0.003	0.053	0.000053

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-15 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.015	0.043	0.001	0.026	0.000011
Geography		By Fuel	0.028	0.084	0.003	0.048	0.000015
		All to INEL	0.044	0.138	0.006	0.078	0.000020
	NTS/SRS	Geographic	0.014	0.040	0.001	0.025	0.000008
		By Fuel	0.027	0.081	0.003	0.047	0.000012
		All to SRS	0.025	0.075	0.003	0.042	0.000011
	NTS/ORR	Geographic	0.015	0.043	0.002	0.027	0.000012
		By Fuel	0.028	0.084	0.003	0.049	0.000016
		All to INEL	0.044	0.138	0.007	0.079	0.000023
		All to SRS	0.026	0.078	0.003	0.045	0.000016
	HS/SRS	Geographic	0.013	0.038	0.001	0.023	0.000008
		By Fuel	0.026	0.079	0.003	0.045	0.000012
		All to SRS	0.024	0.072	0.002	0.040	0.000011
	HS/ORR	Geographic	0.014	0.041	0.001	0.025	0.000012
		By Fuel	0.027	0.082	0.003	0.047	0.000016
		All to INEL	0.043	0.136	0.006	0.077	0.000023
		All to SRS	0.025	0.075	0.002	0.043	0.000016
Centralization	INEL						
	SRS						
	HS	Geographic	0.025	0.074	0.004	0.042	0.000036
		By Fuel	0.038	0.115	0.005	0.063	0.000040
		All to INEL	0.053	0.168	0.008	0.093	0.000027
		All to SRS	0.036	0.109	0.005	0.059	0.000046
	NTS	Geographic	0.024	0.070	0.004	0.042	0.000035
		By Fuel	0.037	0.111	0.006	0.063	0.000040
		All to INEL	0.052	0.164	0.008	0.093	0.000026
		All to SRS	0.035	0.105	0.006	0.059	0.000045
	ORR	Geographic	0.018	0.052	0.002	0.031	0.000016
		By Fuel	0.031	0.093	0.003	0.052	0.000020
		All to INEL	0.048	0.148	0.007	0.083	0.000041
		All to SRS	0.029	0.087	0.003	0.048	0.000017

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-16 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
1992/1993 Planning Basis	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
Regionalization by Fuel Type	INEL/SRS		0.010	0.011	0.014	0.002	0.000005
Regionalization	INEL/SRS		0.006	0.007	0.008	0.001	0.000004
by	INEL/ORR	Geographic	0.008	0.010	0.008	0.002	0.000007
Geography		By Fuel	0.010	0.013	0.012	0.003	0.000008
		All to INEL	0.013	0.013	0.013	0.004	0.000004
	NTS/SRS	Geographic	0.007	0.010	0.008	0.003	0.000005
		By Fuel	0.010	0.013	0.013	0.004	0.000005
		All to SRS	0.009	0.014	0.012	0.004	0.000005
	NTS/ORR	Geographic	0.010	0.016	0.008	0.007	0.000036
		By Fuel	0.011	0.016	0.012	0.005	0.000015
		All to INEL	0.017	0.027	0.015	0.012	0.000039
		All to SRS	0.010	0.014	0.012	0.004	0.000009
	HS/SRS	Geographic	0.006	0.008	0.008	0.001	0.000005
		By Fuel	0.009	0.011	0.012	0.002	0.000005
		All to SRS	0.009	0.011	0.012	0.002	0.000005
	HS/ORR	Geographic	0.009	0.013	0.008	0.005	0.000036
		By Fuel	0.010	0.013	0.012	0.003	0.000009
		All to INEL	0.015	0.022	0.013	0.008	0.000013
		All to SRS	0.009	0.011	0.012	0.002	0.000008
Centralization	INEL		0.013	0.011	0.015	0.003	0.000004
	SRS		0.009	0.011	0.013	0.002	0.000005
	HS	Geographic	0.021	0.048	0.011	0.025	0.000119
		By Fuel	0.013	0.015	0.015	0.004	0.000031
		All to INEL	0.017	0.023	0.016	0.008	0.000013
		All to SRS	0.012	0.014	0.015	0.003	0.000036
	NTS	Geographic	0.020	0.044	0.012	0.025	0.000131
		By Fuel	0.013	0.018	0.016	0.006	0.000035
		All to INEL	0.019	0.028	0.017	0.012	0.000039
		All to SRS	0.013	0.016	0.016	0.005	0.000034
	ORR	Geographic	0.012	0.025	0.008	0.010	0.000041
		By Fuel	0.015	0.029	0.012	0.012	0.000028
		All to INEL	0.027	0.059	0.014	0.031	0.000106
		All to SRS	0.013	0.023	0.012	0.007	0.000009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-17 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
1992/1993 Planning Basis	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
Regionalization by Fuel Type	INEL/SRS		0.048	0.143	0.010	0.088	0.00011
Regionalization	INEL/SRS		0.033	0.096	0.006	0.057	0.00007
by	INEL/ORR	Geographic	0.035	0.101	0.007	0.061	0.00010
Geography		By Fuel	0.046	0.137	0.009	0.085	0.00013
		All to INEL	0.057	0.179	0.011	0.110	0.00012
	NTS/SRS	Geographic	0.034	0.101	0.007	0.060	0.00008
		By Fuel	0.046	0.137	0.010	0.083	0.00011
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	NTS/ORR	Geographic	0.036	0.105	0.007	0.063	0.00011
		By Fuel	0.048	0.142	0.010	0.087	0.00014
		All to INEL	0.062	0.194	0.012	0.118	0.00016
		All to SRS	0.046	0.136	0.010	0.083	0.00014
	HS/SRS	Geographic	0.034	0.098	0.006	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to SRS	0.043	0.127	0.009	0.077	0.00010
	HS/ORR	Geographic	0.035	0.103	0.007	0.061	0.00010
		By Fuel	0.047	0.139	0.009	0.085	0.00013
		All to INEL	0.060	0.189	0.011	0.115	0.00013
		All to SRS	0.045	0.133	0.009	0.081	0.00014
Centralization	INEL		0.065	0.205	0.012	0.126	0.00014
	SRS		0.046	0.137	0.010	0.083	0.00011
	HS	Geographic	0.055	0.169	0.010	0.100	0.00020
		By Fuel	0.067	0.206	0.012	0.124	0.00024
		All to INEL	0.070	0.222	0.013	0.134	0.00015
		All to SRS	0.068	0.209	0.013	0.125	0.00026
	NTS	Geographic	0.054	0.161	0.011	0.100	0.00023
		By Fuel	0.065	0.198	0.013	0.124	0.00026
		All to INEL	0.071	0.222	0.014	0.137	0.00018
		All to SRS	0.066	0.199	0.014	0.125	0.00028
	ORR	Geographic	0.040	0.117	0.007	0.072	0.00013
		By Fuel	0.051	0.152	0.010	0.095	0.00016
		All to INEL	0.071	0.224	0.013	0.139	0.00023
		All to SRS	0.048	0.142	0.010	0.088	0.00014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-18 Tabulation of Overland Transportation Risks: Basic Implementation, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
			Radiological		Nonradiological		Radio-
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.033	0.095	0.006	0.057	0.00007
Geography		By Fuel	0.044	0.131	0.009	0.081	0.00010
		All to INEL	0.057	0.179	0.011	0.110	0.00012
	NTS/SRS	Geographic	0.034	0.098	0.007	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	NTS/ORR	Geographic	0.034	0.097	0.007	0.058	0.00007
		By Fuel	0.045	0.134	0.009	0.082	0.00010
		All to INEL	0.058	0.181	0.011	0.111	0.00013
		All to SRS	0.044	0.129	0.010	0.078	0.00010
	HS/SRS	Geographic	0.033	0.096	0.006	0.057	0.00007
		By Fuel	0.044	0.132	0.009	0.081	0.00010
		All to SRS	0.043	0.127	0.009	0.077	0.00010
	HS/ORR	Geographic	0.033	0.095	0.006	0.057	0.00007
		By Fuel	0.044	0.131	0.009	0.081	0.00010
		All to INEL	0.058	0.179	0.011	0.110	0.00013
		All to SRS	0.043	0.126	0.009	0.077	0.00010
Centralization	INEL						
	SRS						
	HS	Geographic	0.044	0.129	0.009	0.076	0.00011
		By Fuel	0.055	0.165	0.012	0.100	0.00014
		All to INEL	0.068	0.212	0.013	0.129	0.00015
		All to SRS	0.054	0.161	0.012	0.096	0.00015
	NTS	Geographic	0.043	0.126	0.010	0.077	0.00012
		By Fuel	0.054	0.163	0.012	0.101	0.00015
		All to INEL	0.067	0.209	0.013	0.130	0.00015
		All to SRS	0.053	0.158	0.013	0.097	0.00015
	ORR	Geographic	0.036	0.103	0.007	0.063	0.00008
		By Fuel	0.047	0.140	0.010	0.087	0.00011
		All to INEL	0.061	0.188	0.012	0.116	0.00015
		All to SRS	0.046	0.135	0.010	0.083	0.00011

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-19 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
1992/1993 Planning Basis	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
Regionalization by Fuel Type	INEL/SRS		0.013	0.020	0.031	0.003	0.00004
Regionalization by Geography	INEL/SRS		0.010	0.015	0.019	0.002	0.00003
	INEL/ORR	Geographic	0.012	0.019	0.018	0.004	0.00003
		By Fuel	0.014	0.023	0.027	0.005	0.00004
		All to INEL	0.016	0.022	0.041	0.006	0.00005
	NTS/SRS	Geographic	0.012	0.022	0.020	0.005	0.00003
		By Fuel	0.014	0.025	0.028	0.006	0.00004
		All to SRS	0.014	0.026	0.025	0.006	0.00004
	NTS/ORR	Geographic	0.014	0.026	0.019	0.009	0.00006
		By Fuel	0.015	0.027	0.027	0.007	0.00005
		All to INEL	0.021	0.038	0.042	0.013	0.00008
		All to SRS	0.014	0.025	0.024	0.006	0.00004
	HS/SRS	Geographic	0.011	0.019	0.019	0.004	0.00003
		By Fuel	0.014	0.023	0.028	0.005	0.00004
		All to SRS	0.013	0.023	0.025	0.005	0.00004
	HS/ORR	Geographic	0.013	0.024	0.018	0.008	0.00006
		By Fuel	0.014	0.024	0.027	0.006	0.00004
		All to INEL	0.019	0.032	0.041	0.010	0.00006
		All to SRS	0.014	0.023	0.023	0.005	0.00004
Centralization	INEL		0.016	0.020	0.049	0.004	0.00005
	SRS		0.013	0.020	0.027	0.003	0.00004
	HS	Geographic	0.026	0.059	0.030	0.027	0.00015
		By Fuel	0.017	0.027	0.038	0.007	0.00007
		All to INEL	0.021	0.035	0.052	0.011	0.00006
		All to SRS	0.017	0.026	0.036	0.006	0.00007
	NTS	Geographic	0.025	0.056	0.027	0.028	0.00016
		By Fuel	0.018	0.030	0.035	0.008	0.00008
		All to INEL	0.023	0.040	0.049	0.014	0.00009
		All to SRS	0.017	0.028	0.032	0.007	0.00007
	ORR	Geographic	0.016	0.033	0.019	0.014	0.00007
		By Fuel	0.019	0.038	0.028	0.015	0.00007
		All to INEL	0.030	0.067	0.043	0.034	0.00015
		All to SRS	0.016	0.031	0.024	0.011	0.00005

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

shipments. Tables E-11 through E-13 present these risk estimates using average risk parameters. Tables E-14 through E-16 provide the lower bound risk estimates, and Tables E-17 through E-19 provide the upper bound risk estimates.

E.7.3 MEI Results for Routine Conditions

The risks to MEIs under incident-free transportation conditions have been estimated for the exposure scenarios described in Section E.6.7. The estimated dose to each of the receptors considered is presented in Table E-20 on a per-event basis (person-rem per event). Note that the potential exists for individual exposures if multiple exposure events occur. For instance, the dose to a person stuck in traffic next to a spent nuclear fuel shipment for 30 min is calculated to be 11 mrem. If the exposure duration was longer, the dose would rise proportionally. Therefore, it is conceivable that a person could receive a dose on the order of 30 to 50 mrem while stopped in traffic next to a shipment. In addition, a person working at a truck service station could receive a significant dose if trucks were to use the same stops repeatedly. If a truckstop worker was present for 100 shipment stops (at the distance and duration given above), the calculated dose is on the order of 30 mrem. Administrative controls could be instituted to control the location and duration of truck stops if multiple exposures were to happen routinely.

Table E-20 Estimated Doses (Rem/Event) to MEIs During Incident-free Transportation Conditions[a, b]

Receptor		*Dose to MEI*	
		Truck	*Rail*
Workers	Crew Member	0.1 rem/yr[c]	0.1 rem/yr[c]
	Inspector	0.0029 rem/event	0.0029 rem/event
	Rail Yard Crew Member	N/A	1.3×10^{-3} rem/event
Public	Resident	4.0×10^{-7} rem/event	4.0×10^{-7} rem/event
	Person in Traffic Obstruction	0.011 rem/event	0.011 rem/event
	Person at Service Station	0.00031 rem/event	N/A
	Resident Near Rail Stop	N/A	0.000013 rem/event

[a] *The exposure scenario assumptions are described in Section E.6.6.*

[b] *Doses are calculated assuming that the shipment external dose rate is equal to the regulatory limit of 10 mrem per hr at 2 m (6.6 ft) from the shipment.*

[c] *Dose to truck drivers could exceed the legal limit of rem per yr in the absence of administrative controls.*

The cumulative dose to a resident was calculated assuming all 837 shipments arrived at a single port or management site. The cumulative doses assume that the resident is present for every shipment and is unshielded at a distance of 30 m (66 ft) from the route. Therefore, the cumulative dose is only a function of the number of shipments passing a particular point and is independent of the actual site being considered. The maximum dose to this resident, if all the spent nuclear fuel were to be shipped to a single site, would be less than 0.1 mrem. The annual individual dose can be estimated by assuming that shipments would occur uniformly over a 15-year time period.

E.7.4 Accident Consequence Assessment - Maximum Severity Accident Results

The accident consequence assessment is intended to provide an estimate of the maximum potential impacts posed by the most severe potential transportation accidents involving a spent nuclear fuel shipment.

The accident consequence results are presented in Table E-21 for the maximum severity accidents as defined in the modal study. The population doses are for a uniform population density within an 80 km- (50 mi-) radius (Neuhuser and Kanipe, 1993). The location of the MEI is determined based on atmospheric conditions at the time of the accident and the buoyant characteristics of the released plume. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site for neutral and stable conditions, respectively. The dose to the MEI is independent of the location of the accident. In general, the dose to MEIs for the most severe accidents would be less than 10 mrem. No acute or early fatalities would be expected from radiological causes.

Table E-21 Potential Doses to Populations and MEIs for the Most Severe Transportation Accidents Involving Spent Nuclear Fuel[a,b]

	Neutral Conditions[c]				*Stable Conditions[d]*			
	Population[e]		*MEI[f]*		*Population[e]*		*MEI[f]*	
Mode and Accident Location	*Dose (person-rem)*	*Consequences (cancer fatalities)*	*Dose (rem)*	*Consequences (cancer fatality)*	*Dose (person-rem)*	*Consequences (cancer fatalities)*	*Dose (person-rem)*	*Consequences (cancer fatality)*
Truck								
Urban	14	0.007	0.0024	0.0000012	120	0.06	0.0079	0.000004
Suburban	2.7	0.0014	0.0024	0.0000012	21	0.01	0.0079	0.000004
Rural	0.15	0.000075	0.0024	0.0000012	1.2	0.0006	0.0079	0.000004
Rail								
Urban	14	0.007	0.0024	0.0000012	120	0.06	0.0079	0.000004
Suburban	2.7	0.0014	0.0024	0.0000012	21	0.01	0.0079	0.000004
Rural	0.15	0.000075	0.0024	0.0000012	1.2	0.0006	0.0079	0.000004

[a] *The most severe accidents correspond to the modal study accident severity category 6 (DOE, 1995).*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform population density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

The maximum foreseeable offsite transportation accident involves a shipment of spent nuclear fuel in a suburban population zone under neutral (average) weather conditions. The accident has a probability of occurrence of about 1 every 10,000,000 years and could result in 2.7 person-rem and no fatalities. The probability of an accident occurring is at least 10 times smaller in either an urban area or under stable atmospheric conditions, and the consequences are less than 10 times larger.

E.8 Impacts of Implementation Alternatives of the Spent Nuclear Fuel Acceptance Policy

E.8.1 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept only from Developing Nations

This implementation alternative was analyzed using the same set of assumptions as used in analyzing the basic implementation. The results are as follows:

Shipments

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of 168 casks from ports of entry to DOE facilities. The basic shipment count, by point of origin is:

	East Coast		West Coast		
	Aluminum	TRIGA	Aluminum	TRIGA	Totals
Phase 1	31	54	15	30	130
Phase 2	9	16	4	9	38
Totals	**40**	**70**	**19**	**39**	**168**

Calculated in the same manner as described in Section E.7.2.1, the number of intersite shipments for two-phased approaches to this alternative varies between 4 and 33. The variation is caused by the wide variety of phased approaches.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.002 to 0.06 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel, 2) combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.001 to 0.015. The estimated number of radiation-related LCFs for the general population ranged from 0.0006 to 0.045, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0002 to 0.01.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.0000001 to 0.00006 LCFs from radiation and from 0.0001 to 0.028 for traffic fatality, depending on the transportation mode and DOE sites selected. The reasons for the range of fatality estimates are the same as those described for incident-free transportation. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-22 through E-30. The analysis for this alternative implementation is analogous to the analysis performed for the Basic Implementation (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

The consequences of the most severe accident hypothesized are the same as described for the Basic Implementation since the material at risk is the same.

Table E-22 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
1992/1993 Planning Basis	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
Regionalization by Fuel Type	INEL/SRS		0.0098	0.0309	0.0016	0.0182	0.000011
Regionalization by Geography	INEL/SRS		0.0034	0.0106	0.0005	0.0063	0.000003
	INEL/ORR	Geographic	0.0040	0.0120	0.0006	0.0072	0.000009
		By Fuel	0.0087	0.0273	0.0014	0.0161	0.000012
		All to INEL	0.0096	0.0303	0.0015	0.0178	0.000010
	NTS/SRS	Geographic	0.0039	0.0120	0.0007	0.0071	0.000005
		By Fuel	0.0091	0.0285	0.0016	0.0167	0.000014
		All to SRS	0.0070	0.0215	0.0012	0.0127	0.000007
	NTS/ORR	Geographic	0.0044	0.0134	0.0008	0.0080	0.000011
		By Fuel	0.0094	0.0295	0.0017	0.0173	0.000017
		All to INEL	0.0106	0.0335	0.0018	0.0195	0.000018
		All to SRS	0.0077	0.0234	0.0014	0.0140	0.000016
	HS/SRS	Geographic	0.0035	0.0109	0.0005	0.0064	0.000003
		By Fuel	0.0086	0.0272	0.0013	0.0159	0.000010
		All to SRS	0.0067	0.0205	0.0011	0.0122	0.000007
	HS/ORR	Geographic	0.0040	0.0122	0.0006	0.0073	0.000009
		By Fuel	0.0090	0.0281	0.0014	0.0165	0.000013
		All to INEL	0.0101	0.0319	0.0015	0.0185	0.000012
		All to SRS	0.0074	0.0224	0.0012	0.0134	0.000016
Centralization	INEL		0.0112	0.0356	0.0017	0.0208	0.000012
	SRS		0.0078	0.0241	0.0013	0.0142	0.000008
	HS	Geographic	0.0078	0.0245	0.0011	0.0142	0.000025
		By Fuel	0.0119	0.0377	0.0018	0.0219	0.000023
		All to INEL	0.0120	0.0384	0.0018	0.0221	0.000014
		All to SRS	0.0120	0.0375	0.0019	0.0220	0.000039
	NTS	Geographic	0.0077	0.0237	0.0014	0.0146	0.000030
		By Fuel	0.0119	0.0376	0.0021	0.0224	0.000028
		All to INEL	0.0123	0.0390	0.0022	0.0229	0.000020
		All to SRS	0.0117	0.0361	0.0020	0.0221	0.000043
	ORR	Geographic	0.0056	0.0173	0.0008	0.0105	0.000017
		By Fuel	0.0110	0.0347	0.0017	0.0207	0.000026
		All to INEL	0.0127	0.0404	0.0019	0.0240	0.000032
		All to SRS	0.0083	0.0255	0.0014	0.0153	0.000017

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-23 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by Geography	INEL/ORR	Geographic	0.0037	0.0111	0.0006	0.0065	0.000004
		By Fuel	0.0086	0.0268	0.0014	0.0157	0.000009
		All to INEL	0.0096	0.0303	0.0015	0.0178	0.000010
	NTS/SRS	Geographic	0.0036	0.0111	0.0006	0.0066	0.000003
		By Fuel	0.0086	0.0269	0.0015	0.0158	0.000009
		All to SRS	0.0070	0.0215	0.0012	0.0127	0.000007
	NTS/ORR	Geographic	0.0039	0.0115	0.0007	0.0068	0.000004
		By Fuel	0.0088	0.0273	0.0015	0.0160	0.000010
		All to INEL	0.0098	0.0308	0.0016	0.0181	0.000011
		All to SRS	0.0072	0.0220	0.0013	0.0130	0.000009
	HS/SRS	Geographic	0.0033	0.0101	0.0005	0.0060	0.000003
		By Fuel	0.0083	0.0258	0.0013	0.0152	0.000009
		All to SRS	0.0067	0.0205	0.0011	0.0122	0.000007
	HS/ORR	Geographic	0.0035	0.0105	0.0006	0.0063	0.000004
		By Fuel	0.0085	0.0263	0.0014	0.0155	0.000010
		All to INEL	0.0095	0.0298	0.0015	0.0175	0.000011
		All to SRS	0.0069	0.0210	0.0012	0.0125	0.000008
Centralization	INEL						
	SRS						
	HS	Geographic	0.0057	0.0171	0.0010	0.0099	0.000009
		By Fuel	0.0105	0.0328	0.0018	0.0191	0.000014
		All to INEL	0.0115	0.0363	0.0018	0.0211	0.000013
		All to SRS	0.0091	0.0277	0.0017	0.0161	0.000016
	NTS	Geographic	0.0057	0.0172	0.0012	0.0103	0.000010
		By Fuel	0.0106	0.0329	0.0019	0.0195	0.000014
		All to INEL	0.0115	0.0364	0.0019	0.0215	0.000013
		All to SRS	0.0091	0.0277	0.0019	0.0165	0.000016
	ORR	Geographic	0.0045	0.0137	0.0008	0.0082	0.000006
		By Fuel	0.0095	0.0295	0.0016	0.0174	0.000013
		All to INEL	0.0106	0.0331	0.0018	0.0194	0.000015
		All to SRS	0.0079	0.0241	0.0013	0.0143	0.000009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-24 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
1992/1993 Planning Basis	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
Regionalization by Fuel Type	INEL/SRS		0.0027	0.0027	0.0034	0.0006	0.000003
Regionalization	INEL/SRS		0.0015	0.0011	0.0010	0.0002	0.000001
by	INEL/ORR	Geographic	0.0018	0.0023	0.0010	0.0008	0.000001
Geography		By Fuel	0.0028	0.0035	0.0029	0.0011	0.000003
		All to INEL	0.0029	0.0036	0.0031	0.0011	0.000003
	NTS/SRS	Geographic	0.0019	0.0027	0.0011	0.0010	0.000001
		By Fuel	0.0029	0.0039	0.0030	0.0014	0.000003
		All to SRS	0.0025	0.0038	0.0025	0.0013	0.000002
	NTS/ORR	Geographic	0.0023	0.0037	0.0011	0.0017	0.000007
		By Fuel	0.0035	0.0057	0.0031	0.0023	0.000008
		All to INEL	0.0039	0.0068	0.0035	0.0028	0.000010
		All to SRS	0.0027	0.0039	0.0026	0.0013	0.000003
	HS/SRS	Geographic	0.0016	0.0017	0.0009	0.0005	0.000001
		By Fuel	0.0026	0.0029	0.0028	0.0008	0.000003
		All to SRS	0.0022	0.0028	0.0024	0.0007	0.000002
	HS/ORR	Geographic	0.0020	0.0027	0.0010	0.0011	0.000007
		By Fuel	0.0030	0.0044	0.0029	0.0014	0.000004
		All to INEL	0.0034	0.0052	0.0032	0.0019	0.000004
		All to SRS	0.0024	0.0029	0.0025	0.0007	0.000003
Centralization	INEL		0.0030	0.0028	0.0038	0.0007	0.000003
	SRS		0.0024	0.0026	0.0029	0.0005	0.000002
	HS	Geographic	0.0042	0.0089	0.0020	0.0046	0.000022
		By Fuel	0.0035	0.0048	0.0037	0.0016	0.000006
		All to INEL	0.0038	0.0056	0.0039	0.0020	0.000005
		All to SRS	0.0030	0.0036	0.0035	0.0009	0.000009
	NTS	Geographic	0.0042	0.0089	0.0020	0.0050	0.000024
		By Fuel	0.0039	0.0062	0.0039	0.0024	0.000010
		All to INEL	0.0043	0.0072	0.0041	0.0030	0.000010
		All to SRS	0.0033	0.0045	0.0035	0.0015	0.000009
	ORR	Geographic	0.0030	0.0059	0.0012	0.0030	0.000009
		By Fuel	0.0050	0.0109	0.0031	0.0056	0.000017
		All to INEL	0.0060	0.0137	0.0035	0.0074	0.000024
		All to SRS	0.0033	0.0061	0.0026	0.0026	0.000004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-25 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipment via Truck, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
1992/1993 Planning Basis	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
Regionalization by Fuel Type	INEL/SRS		0.0080	0.0256	0.0010	0.0142	0.0000022
Regionalization	INEL/SRS		0.0023	0.0072	0.0002	0.0039	0.0000004
by	INEL/ORR	Geographic	0.0029	0.0087	0.0003	0.0050	0.0000061
Geography		By Fuel	0.0071	0.0226	0.0009	0.0127	0.0000049
		All to INEL	0.0077	0.0245	0.0011	0.0141	0.0000020
	NTS/SRS	Geographic	0.0026	0.0082	0.0004	0.0046	0.0000031
		By Fuel	0.0073	0.0232	0.0011	0.0130	0.0000064
		All to SRS	0.0058	0.0181	0.0007	0.0096	0.0000018
	NTS/ORR	Geographic	0.0032	0.0098	0.0005	0.0058	0.0000088
		By Fuel	0.0077	0.0244	0.0012	0.0139	0.0000096
		All to INEL	0.0086	0.0273	0.0014	0.0157	0.0000093
		All to SRS	0.0066	0.0202	0.0008	0.0111	0.0000103
	HS/SRS	Geographic	0.0023	0.0073	0.0002	0.0039	0.0000010
		By Fuel	0.0069	0.0221	0.0009	0.0122	0.0000030
		All to SRS	0.0056	0.0173	0.0006	0.0091	0.0000017
	HS/ORR	Geographic	0.0029	0.0088	0.0003	0.0051	0.0000068
		By Fuel	0.0074	0.0233	0.0010	0.0130	0.0000062
		All to INEL	0.0081	0.0260	0.0011	0.0147	0.0000039
		All to SRS	0.0063	0.0194	0.0007	0.0105	0.0000102
Centralization	INEL		0.0090	0.0288	0.0013	0.0164	0.0000023
	SRS		0.0068	0.0212	0.0007	0.0110	0.0000021
	HS	Geographic	0.0063	0.0201	0.0008	0.0112	0.0000215
		By Fuel	0.0100	0.0319	0.0013	0.0177	0.0000143
		All to INEL	0.0098	0.0315	0.0014	0.0176	0.0000043
		All to SRS	0.0106	0.0336	0.0013	0.0184	0.0000320
	NTS	Geographic	0.0060	0.0188	0.0010	0.0113	0.0000255
		By Fuel	0.0098	0.0312	0.0016	0.0180	0.0000189
		All to INEL	0.0098	0.0316	0.0017	0.0181	0.0000096
		All to SRS	0.0101	0.0316	0.0015	0.0183	0.0000352
	ORR	Geographic	0.0045	0.0142	0.0005	0.0080	0.0000140
		By Fuel	0.0095	0.0301	0.0012	0.0170	0.0000185
		All to INEL	0.0108	0.0348	0.0014	0.0200	0.0000232
		All to SRS	0.0074	0.0229	0.0008	0.0122	0.0000105

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-26 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.0025	0.0078	0.0003	0.0044	0.0000011
Geography		By Fuel	0.0070	0.0221	0.0009	0.0124	0.0000022
		All to INEL	0.0077	0.0245	0.0011	0.0141	0.0000020
	NTS/SRS	Geographic	0.0023	0.0072	0.0004	0.0041	0.0000007
		By Fuel	0.0068	0.0215	0.0010	0.0121	0.0000022
		All to SRS	0.0058	0.0181	0.0007	0.0096	0.0000018
	NTS/ORR	Geographic	0.0026	0.0079	0.0004	0.0046	0.0000014
		By Fuel	0.0071	0.0222	0.0010	0.0126	0.0000027
		All to INEL	0.0078	0.0247	0.0012	0.0143	0.0000027
		All to SRS	0.0061	0.0188	0.0008	0.0101	0.0000028
	HS/SRS	Geographic	0.0021	0.0065	0.0002	0.0035	0.0000006
		By Fuel	0.0066	0.0208	0.0009	0.0115	0.0000023
		All to SRS	0.0056	0.0173	0.0006	0.0091	0.0000017
	HS/ORR	Geographic	0.0024	0.0071	0.0003	0.0041	0.0000014
		By Fuel	0.0069	0.0214	0.0009	0.0121	0.0000027
		All to INEL	0.0076	0.0239	0.0011	0.0137	0.0000028
		All to SRS	0.0059	0.0180	0.0006	0.0096	0.0000026
Centralization	INEL						
	SRS						
	HS	Geographic	0.0042	0.0128	0.0007	0.0069	0.0000054
		By Fuel	0.0086	0.0270	0.0013	0.0149	0.0000051
		All to INEL	0.0092	0.0294	0.0013	0.0166	0.0000031
		All to SRS	0.0077	0.0237	0.0011	0.0125	0.0000083
	NTS	Geographic	0.0040	0.0123	0.0008	0.0071	0.0000052
		By Fuel	0.0084	0.0265	0.0014	0.0150	0.0000050
		All to INEL	0.0091	0.0289	0.0015	0.0167	0.0000030
		All to SRS	0.0076	0.0232	0.0013	0.0126	0.0000081
	ORR	Geographic	0.0035	0.0106	0.0004	0.0057	0.0000029
		By Fuel	0.0080	0.0249	0.0011	0.0137	0.0000053
		All to INEL	0.0088	0.0274	0.0013	0.0154	0.0000063
		All to SRS	0.0069	0.0214	0.0007	0.0112	0.0000030

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-27 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
1992/1993 Planning Basis	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
Regionalization by Fuel Type	INEL/SRS		0.0023	0.0017	0.0021	0.0005	0.0000003
Regionalization by Geography	INEL/SRS		0.0012	0.0006	0.0004	0.0001	0.0000001
	INEL/ORR	Geographic	0.0015	0.0018	0.0004	0.0006	0.0000007
		By Fuel	0.0024	0.0026	0.0017	0.0009	0.0000006
		All to INEL	0.0024	0.0025	0.0018	0.0009	0.0000002
	NTS/SRS	Geographic	0.0015	0.0017	0.0005	0.0008	0.0000004
		By Fuel	0.0024	0.0026	0.0018	0.0011	0.0000007
		All to SRS	0.0021	0.0027	0.0018	0.0010	0.0000005
	NTS/ORR	Geographic	0.0019	0.0028	0.0006	0.0015	0.0000060
		By Fuel	0.0029	0.0044	0.0020	0.0020	0.0000053
		All to INEL	0.0033	0.0053	0.0022	0.0026	0.0000075
		All to SRS	0.0023	0.0029	0.0019	0.0011	0.0000013
	HS/SRS	Geographic	0.0012	0.0010	0.0004	0.0002	0.0000004
		By Fuel	0.0021	0.0018	0.0017	0.0005	0.0000008
		All to SRS	0.0019	0.0020	0.0016	0.0005	0.0000003
	HS/ORR	Geographic	0.0017	0.0020	0.0005	0.0009	0.0000060
		By Fuel	0.0026	0.0033	0.0018	0.0012	0.0000018
		All to INEL	0.0029	0.0040	0.0019	0.0016	0.0000022
		All to SRS	0.0021	0.0021	0.0017	0.0005	0.0000012
Centralization	INEL		0.0025	0.0017	0.0021	0.0005	0.0000002
	SRS		0.0021	0.0020	0.0021	0.0004	0.0000004
	HS	Geographic	0.0037	0.0080	0.0011	0.0043	0.0000204
		By Fuel	0.0030	0.0036	0.0023	0.0013	0.0000039
		All to INEL	0.0032	0.0042	0.0022	0.0017	0.0000022
		All to SRS	0.0026	0.0026	0.0024	0.0007	0.0000066
	NTS	Geographic	0.0037	0.0077	0.0013	0.0047	0.0000224
		By Fuel	0.0033	0.0046	0.0025	0.0021	0.0000072
		All to INEL	0.0036	0.0054	0.0026	0.0026	0.0000075
		All to SRS	0.0027	0.0032	0.0025	0.0012	0.0000064
	ORR	Geographic	0.0028	0.0055	0.0006	0.0026	0.0000075
		By Fuel	0.0047	0.0101	0.0020	0.0052	0.0000142
		All to INEL	0.0056	0.0128	0.0021	0.0069	0.0000214
		All to SRS	0.0031	0.0056	0.0018	0.0022	0.0000015

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-28 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
1992/1993 Planning Basis	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
Regionalization by Fuel Type	INEL/SRS		0.0113	0.0352	0.0024	0.0220	0.000027
Regionalization by Geography	INEL/SRS		0.0064	0.0191	0.0014	0.0111	0.000014
	INEL/ORR	Geographic	0.0067	0.0200	0.0015	0.0117	0.000019
		By Fuel	0.0104	0.0320	0.0022	0.0199	0.000027
		All to INEL	0.0107	0.0335	0.0021	0.0211	0.000024
	NTS/SRS	Geographic	0.0069	0.0207	0.0016	0.0118	0.000017
		By Fuel	0.0110	0.0340	0.0025	0.0207	0.000029
		All to SRS	0.0099	0.0296	0.0024	0.0176	0.000024
	NTS/ORR	Geographic	0.0072	0.0215	0.0016	0.0125	0.000022
		By Fuel	0.0111	0.0344	0.0025	0.0210	0.000032
		All to INEL	0.0117	0.0368	0.0025	0.0227	0.000032
		All to SRS	0.0104	0.0310	0.0025	0.0186	0.000032
	HS/SRS	Geographic	0.0067	0.0198	0.0014	0.0113	0.000015
		By Fuel	0.0106	0.0329	0.0022	0.0201	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000023
	HS/ORR	Geographic	0.0070	0.0206	0.0015	0.0120	0.000020
		By Fuel	0.0108	0.0333	0.0022	0.0204	0.000028
		All to INEL	0.0113	0.0355	0.0022	0.0220	0.000026
		All to SRS	0.0102	0.0303	0.0024	0.0183	0.000031
Centralization	INEL		0.0120	0.0379	0.0024	0.0240	0.000028
	SRS		0.0106	0.0319	0.0026	0.0192	0.000026
	HS	Geographic	0.0104	0.0320	0.0020	0.0188	0.000038
		By Fuel	0.0134	0.0420	0.0026	0.0257	0.000040
		All to INEL	0.0130	0.0411	0.0025	0.0254	0.000030
		All to SRS	0.0145	0.0445	0.0030	0.0267	0.000057
	NTS	Geographic	0.0102	0.0310	0.0022	0.0189	0.000043
		By Fuel	0.0134	0.0417	0.0029	0.0260	0.000045
		All to INEL	0.0131	0.0415	0.0028	0.0260	0.000036
		All to SRS	0.0141	0.0428	0.0032	0.0267	0.000060
	ORR	Geographic	0.0083	0.0251	0.0017	0.0152	0.000029
		By Fuel	0.0126	0.0393	0.0026	0.0247	0.000042
		All to INEL	0.0137	0.0435	0.0026	0.0275	0.000047
		All to SRS	0.0109	0.0328	0.0026	0.0202	0.000033

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-29 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.0064	0.0190	0.0014	0.0111	0.000014
Geography		By Fuel	0.0102	0.0315	0.0022	0.0195	0.000024
		All to INEL	0.0107	0.0335	0.0021	0.0211	0.000024
	NTS/SRS	Geographic	0.0067	0.0197	0.0015	0.0113	0.000015
		By Fuel	0.0105	0.0323	0.0023	0.0198	0.000025
		All to SRS	0.0099	0.0296	0.0024	0.0176	0.000024
	NTS/ORR	Geographic	0.0067	0.0196	0.0015	0.0113	0.000015
		By Fuel	0.0105	0.0322	0.0023	0.0198	0.000025
		All to INEL	0.0110	0.0341	0.0023	0.0213	0.000025
		All to SRS	0.0099	0.0296	0.0024	0.0177	0.000024
	HS/SRS	Geographic	0.0065	0.0190	0.0014	0.0110	0.000015
		By Fuel	0.0103	0.0316	0.0022	0.0194	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000023
	HS/ORR	Geographic	0.0065	0.0189	0.0014	0.0110	0.000015
		By Fuel	0.0103	0.0314	0.0022	0.0195	0.000025
		All to INEL	0.0108	0.0334	0.0022	0.0210	0.000025
		All to SRS	0.0097	0.0289	0.0023	0.0173	0.000024
Centralization	INEL						
	SRS						
	HS	Geographic	0.0083	0.0246	0.0019	0.0145	0.000022
		By Fuel	0.0120	0.0371	0.0026	0.0229	0.000030
		All to INEL	0.0124	0.0390	0.0024	0.0244	0.000029
		All to SRS	0.0116	0.0346	0.0029	0.0209	0.000033
	NTS	Geographic	0.0082	0.0245	0.0020	0.0146	0.000023
		By Fuel	0.0120	0.0370	0.0027	0.0231	0.000031
		All to INEL	0.0124	0.0388	0.0026	0.0246	0.000030
		All to SRS	0.0115	0.0345	0.0030	0.0210	0.000033
	ORR	Geographic	0.0073	0.0215	0.0017	0.0129	0.000018
		By Fuel	0.0111	0.0341	0.0025	0.0214	0.000029
		All to INEL	0.0116	0.0361	0.0025	0.0229	0.000030
		All to SRS	0.0104	0.0314	0.0025	0.0192	0.000026

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-30 Tabulation of Overland Transportation Risks: Spent Nuclear Fuel from Developing Nations Only, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine				Accidental
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
1992/1993 Planning Basis	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
Regionalization by Fuel Type	INEL/SRS		0.0030	0.0036	0.0076	0.0007	0.000010
Regionalization by Geography	INEL/SRS		0.0020	0.0025	0.0028	0.0004	0.000005
	INEL/ORR	Geographic	0.0024	0.0037	0.0027	0.0011	0.000006
		By Fuel	0.0031	0.0046	0.0064	0.0013	0.000009
		All to INEL	0.0031	0.0044	0.0076	0.0013	0.000009
	NTS/SRS	Geographic	0.0026	0.0044	0.0029	0.0013	0.000006
		By Fuel	0.0034	0.0053	0.0067	0.0016	0.000010
		All to SRS	0.0032	0.0056	0.0046	0.0015	0.000009
	NTS/ORR	Geographic	0.0029	0.0053	0.0028	0.0019	0.000011
		By Fuel	0.0038	0.0069	0.0067	0.0025	0.000014
		All to INEL	0.0042	0.0078	0.0080	0.0030	0.000017
		All to SRS	0.0033	0.0056	0.0045	0.0015	0.000009
	HS/SRS	Geographic	0.0023	0.0037	0.0028	0.0010	0.000006
		By Fuel	0.0031	0.0046	0.0065	0.0012	0.000010
		All to SRS	0.0030	0.0049	0.0045	0.0012	0.000008
	HS/ORR	Geographic	0.0027	0.0046	0.0027	0.0016	0.000011
		By Fuel	0.0035	0.0059	0.0064	0.0019	0.000011
		All to INEL	0.0038	0.0065	0.0077	0.0023	0.000012
		All to SRS	0.0031	0.0048	0.0044	0.0012	0.000009
Centralization	INEL		0.0031	0.0035	0.0092	0.0007	0.000010
	SRS		0.0029	0.0041	0.0051	0.0007	0.000009
	HS	Geographic	0.0048	0.0107	0.0050	0.0051	0.000027
		By Fuel	0.0039	0.0063	0.0086	0.0020	0.000014
		All to INEL	0.0041	0.0069	0.0098	0.0024	0.000013
		All to SRS	0.0036	0.0055	0.0067	0.0014	0.000016
	NTS	Geographic	0.0048	0.0104	0.0043	0.0052	0.000029
		By Fuel	0.0043	0.0074	0.0080	0.0026	0.000018
		All to INEL	0.0045	0.0082	0.0092	0.0031	0.000019
		All to SRS	0.0038	0.0061	0.0060	0.0017	0.000016
	ORR	Geographic	0.0035	0.0072	0.0030	0.0036	0.000014
		By Fuel	0.0053	0.0118	0.0067	0.0061	0.000024
		All to INEL	0.0062	0.0144	0.0081	0.0078	0.000032
		All to SRS	0.0038	0.0074	0.0046	0.0031	0.000011

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

E.8.2 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept Only from Reactors that Use Highly-Enriched Uranium (HEU)

This alternative was not analyzed for policy reasons. See Chapter 4.

E.8.3 Implementation Alternative - Implementing an Acceptance Policy of Alternative Amounts of Spent Nuclear Fuel - Accept Target Material

Target material is currently stored overseas as a liquid. In order to allow shipment, it must be processed into a solid form by either calcination or oxidation. Calcination results in a solid, but easily crumbled material, and oxidation results in a powder. Oxidation removes the aluminum and, therefore, would lead to fewer shipments than calcination. Shipment counts in Appendix B indicate that just over five shipments would be arriving on the east coast. However, in order to be conservative, six full shipments are used for transportation risk analysis. Similarly, the amount of material that could arrive on the west coast is much less than one full cask. The analysis conservatively assumes one full cask.

Form	Port of Entry		
	East Coast	West Coast	Eastern Canada
Calcinate	14	1	125
Oxidized Powder	6	1	50

Analysis of the target material and potential casks indicates that the maximum dose rate from any cask would be 0.1 mrem per hr at 2 m (3.3 ft). This low radiation level is based on the low burn-up of target material. Because of the conservative release fractions assigned to the oxidized material (see Section E.6.4.2), the results are emphasized below. The risks tabulated in this section would be added to those associated with the basic implementation of Management Alternative 1 if both aspects of the policy were to be performed.

Impacts of Incident-Free Ground Transport

The incident-free transportation of oxidized target material was estimated to result in total latent fatalities that ranged from 0.0002 to 0.003 over the entire duration of the program. The calcinated target material results are 2.5 times higher. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew. This represents an increase to the risk associated with the basic implementation.

The range of fatality estimates is caused by three factors: 1) the option of using truck or rail to transport spent nuclear fuel 2) combinations of Phase 1 and Phase 2 sites that created varying shipment numbers and distances, and 3) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.00007 to 0.00074. The estimated number of radiation-related LCFs for the general population ranged from 0.00015 to 0.0023, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.00396.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire policy are estimated to range from 0.00023 to 0.0054 LCF from radiation and from 0.0001 to 0.013 for traffic fatality, depending on the transportation mode and DOE sites selected. The risks would be four times lower if calcinated material is transported. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-31 through E-39. The analysis for this implementation alternative is analogous to the analysis performed for the Basic Implementation (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2. The total policy risk with this implementation alternative is the sum of the values in the above referenced tables and those in Section E.7.2 describing the Basic Implementation.

Table E-40 gives the consequences for the most severe accident hypothesized if that accident were to occur at various locations. The maximum accident risks would be four times lower for calcinated material. The accident probabilities are described in Section 6 of this appendix.

E.8.4 Implementation Alternative - Implementing an Acceptance Policy for Varying Durations - Five-Year Spent Nuclear Fuel Acceptance

Under all SNF&INEL Final EIS (DOE, 1995) alternatives, the shipment of foreign research reactor spent nuclear fuel would require the movement of casks from ports of entry to DOE facilities. The basic shipment count, by point of origin is:

	East Coast		*West Coast*		Totals
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	
Phase 1	419	101	105	53	678

Calculated in the same manner as described for the basic implementation of Management Alternative 1, the number of intersite shipments for the two-phased approaches to this strategy varies between 8 and 184. The variation is caused by the wide variety of phased approaches.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.01 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.005 to 0.064. The estimated number of radiation-related LCFs for the general population ranged from 0.005 to 0.20, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.001 to 0.041.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000003 to 0.00026 LCFs from radiation and from 0.001 to 0.13 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-41 through E-49. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

Table E-31 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio- logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
1992/1993 Planning Basis	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
Regionalization by Fuel Type	INEL/SRS		0.00023	0.00064	0.00016	0.0041	0.00074
Regionalization	INEL/SRS		0.00022	0.00061	0.00013	0.0040	0.00070
by	INEL/ORR	Geographic	0.00029	0.00079	0.00026	0.0051	0.00071
Geography		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00042	0.00129	0.00060	0.0066	0.00352
	NTS/SRS	Geographic	0.00022	0.00062	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	NTS/ORR	Geographic	0.00029	0.00080	0.00027	0.0052	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00054	0.00166	0.00095	0.0085	0.00353
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075
	HS/SRS	Geographic	0.00022	0.00062	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	HS/ORR	Geographic	0.00029	0.00080	0.00027	0.0051	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00051	0.00159	0.00069	0.0079	0.00352
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075
Centralization	INEL		0.00049	0.00152	0.00074	0.0075	0.00439
	SRS		0.00023	0.00055	0.00016	0.0041	0.00074
	HS	Geographic	0.00072	0.00221	0.00087	0.0128	0.00163
		By Fuel	0.00073	0.00225	0.00089	0.0131	0.00167
		All to INEL	0.00059	0.00186	0.00086	0.0091	0.00442
		All to SRS	0.00073	0.00225	0.00089	0.0131	0.00167
	NTS	Geographic	0.00067	0.00199	0.00094	0.0127	0.00160
		By Fuel	0.00068	0.00203	0.00096	0.0129	0.00165
		All to INEL	0.00061	0.00191	0.00114	0.0098	0.00439
		All to SRS	0.00068	0.00203	0.00096	0.0129	0.00165
	ORR	Geographic	0.00030	0.00082	0.00027	0.0053	0.00071
		By Fuel	0.00030	0.00082	0.00028	0.0053	0.00075
		All to INEL	0.00073	0.00230	0.00095	0.0128	0.00355
		All to SRS	0.00030	0.00082	0.00028	0.0053	0.00075

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-32 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00042	0.00129	0.00060	0.0066	0.00352
	NTS/SRS	Geographic	0.00022	0.00061	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	NTS/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00047	0.00132	0.00077	0.0066	0.00352
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074
	HS/SRS	Geographic	0.00022	0.00061	0.00015	0.0040	0.00070
		By Fuel	0.00023	0.00064	0.00016	0.0041	0.00074
		All to SRS	0.00023	0.00064	0.00016	0.0041	0.00074
	HS/ORR	Geographic	0.00025	0.00062	0.00025	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00046	0.00132	0.00077	0.0066	0.00352
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074
Centralization	INEL						
	SRS						
	HS	Geographic	0.00041	0.00097	0.00125	0.0053	0.00161
		By Fuel	0.00041	0.00100	0.00129	0.0054	0.00165
		All to INEL	0.00054	0.00159	0.00094	0.0078	0.00442
		All to SRS	0.00041	0.00100	0.00129	0.0054	0.00165
	NTS	Geographic	0.00040	0.00094	0.00130	0.0054	0.00158
		By Fuel	0.00041	0.00097	0.00133	0.0055	0.00162
		All to INEL	0.00054	0.00157	0.00096	0.0079	0.00439
		All to SRS	0.00041	0.00097	0.00133	0.0055	0.00162
	ORR	Geographic	0.00025	0.00063	0.00026	0.0039	0.00070
		By Fuel	0.00026	0.00065	0.00026	0.0040	0.00074
		All to INEL	0.00050	0.00136	0.00115	0.0068	0.00353
		All to SRS	0.00026	0.00065	0.00026	0.0040	0.00074

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-33 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Average Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
			Radiological		Nonradiological		Radio-
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	logical
Decentralization	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
1992/1993 Planning Basis	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
Regionalization by Fuel Type	INEL/SRS		0.00007	0.00019	0.00295	0.0001	0.00024
Regionalization	INEL/SRS		0.00007	0.00018	0.00228	0.0001	0.00023
by	INEL/ORR	Geographic	0.00010	0.00019	0.00267	0.0002	0.00021
Geography		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00010	0.00016	0.00217	0.0002	0.00085
	NTS/SRS	Geographic	0.00007	0.00018	0.00292	0.0001	0.00023
		By Fuel	0.00007	0.00019	0.00295	0.0001	0.00024
		All to SRS	0.00007	0.00019	0.00295	0.0001	0.00024
	NTS/ORR	Geographic	0.00014	0.00036	0.00269	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00022	0.00052	0.00253	0.0022	0.00086
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022
	HS/SRS	Geographic	0.00007	0.00018	0.00292	0.0001	0.00023
		By Fuel	0.00007	0.00019	0.00295	0.0001	0.00024
		All to SRS	0.00007	0.00019	0.00295	0.0001	0.00024
	HS/ORR	Geographic	0.00014	0.00036	0.00269	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00019	0.00045	0.00227	0.0016	0.00085
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022
Centralization	INEL		0.00012	0.00016	0.00239	0.0003	0.00106
	SRS		0.00007	0.00015	0.00295	0.0001	0.00024
	HS	Geographic	0.00050	0.00151	0.00337	0.0080	0.00047
		By Fuel	0.00020	0.00029	0.00381	0.0005	0.00046
		All to INEL	0.00021	0.00047	0.00251	0.0016	0.00107
		All to SRS	0.00020	0.00029	0.00381	0.0005	0.00046
	NTS	Geographic	0.00046	0.00131	0.00343	0.0077	0.00047
		By Fuel	0.00019	0.00028	0.00384	0.0004	0.00046
		All to INEL	0.00024	0.00054	0.00278	0.0022	0.00107
		All to SRS	0.00019	0.00028	0.00384	0.0004	0.00046
	ORR	Geographic	0.00014	0.00036	0.00270	0.0014	0.00022
		By Fuel	0.00010	0.00019	0.00270	0.0002	0.00022
		All to INEL	0.00041	0.00117	0.00253	0.0064	0.00087
		All to SRS	0.00010	0.00019	0.00270	0.0002	0.00022

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-34 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
1992/1993 Planning Basis	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
Regionalization by Fuel Type	INEL/SRS		0.00022	0.00062	0.00014	0.0040	0.00070
Regionalization by Geography	INEL/SRS		0.00021	0.00060	0.00013	0.0039	0.00068
	INEL/ORR	Geographic	0.00028	0.00078	0.00025	0.0050	0.00095
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00041	0.00126	0.00058	0.0064	0.00341
	NTS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00069
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	NTS/ORR	Geographic	0.00029	0.00079	0.00026	0.0050	0.00096
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00053	0.00163	0.00093	0.0083	0.00358
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098
	HS/SRS	Geographic	0.00022	0.00060	0.00013	0.0039	0.00069
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	HS/ORR	Geographic	0.00029	0.00079	0.00025	0.0050	0.00096
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00050	0.00156	0.00067	0.0077	0.00356
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098
Centralization	INEL		0.00048	0.00147	0.00072	0.0073	0.00426
	SRS		0.00022	0.00062	0.00014	0.0040	0.00070
	HS	Geographic	0.00071	0.00219	0.00085	0.0127	0.00369
		By Fuel	0.00073	0.00224	0.00087	0.0130	0.00375
		All to INEL	0.00058	0.00183	0.00084	0.0089	0.00443
		All to SRS	0.00073	0.00224	0.00087	0.0130	0.00375
	NTS	Geographic	0.00066	0.00197	0.00093	0.0125	0.00362
		By Fuel	0.00068	0.00201	0.00094	0.0128	0.00369
		All to INEL	0.00060	0.00188	0.00111	0.0095	0.00442
		All to SRS	0.00068	0.00201	0.00094	0.0128	0.00369
	ORR	Geographic	0.00029	0.00080	0.00026	0.0051	0.00100
		By Fuel	0.00029	0.00081	0.00026	0.0052	0.00098
		All to INEL	0.00072	0.00227	0.00093	0.0126	0.00528
		All to SRS	0.00029	0.00081	0.00026	0.0052	0.00098

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-35 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite via Rail, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.00024	0.00061	0.00023	0.0038	0.00071
Geography		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00041	0.00126	0.00058	0.0064	0.00341
	NTS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00068
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	NTS/ORR	Geographic	0.00025	0.00061	0.00024	0.0038	0.00071
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00046	0.00129	0.00075	0.0065	0.00343
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073
	HS/SRS	Geographic	0.00022	0.00060	0.00014	0.0039	0.00068
		By Fuel	0.00022	0.00062	0.00014	0.0040	0.00070
		All to SRS	0.00022	0.00062	0.00014	0.0040	0.00070
	HS/ORR	Geographic	0.00025	0.00061	0.00024	0.0038	0.00071
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00045	0.00129	0.00075	0.0064	0.00344
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073
Centralization	INEL						
	SRS						
	HS	Geographic	0.00040	0.00096	0.00124	0.0052	0.00194
		By Fuel	0.00041	0.00098	0.00127	0.0053	0.00197
		All to INEL	0.00053	0.00156	0.00091	0.0076	0.00432
		All to SRS	0.00041	0.00098	0.00127	0.0053	0.00197
	NTS	Geographic	0.00039	0.00092	0.00129	0.0052	0.00188
		By Fuel	0.00040	0.00095	0.00131	0.0054	0.00190
		All to INEL	0.00053	0.00154	0.00093	0.0077	0.00426
		All to SRS	0.00040	0.00095	0.00131	0.0054	0.00190
	ORR	Geographic	0.00025	0.00061	0.00025	0.0038	0.00072
		By Fuel	0.00025	0.00063	0.00024	0.0039	0.00073
		All to INEL	0.00049	0.00132	0.00113	0.0066	0.00376
		All to SRS	0.00025	0.00063	0.00024	0.0039	0.00073

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-36 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Lower Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
1992/1993 Planning Basis	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
Regionalization by Fuel Type	INEL/SRS		0.00007	0.00018	0.00292	0.0001	0.00023
Regionalization by Geography	INEL/SRS		0.00007	0.00018	0.00289	0.0001	0.00023
	INEL/ORR	Geographic	0.00010	0.00019	0.00264	0.0002	0.00023
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00010	0.00015	0.00210	0.0002	0.00080
	NTS/SRS	Geographic	0.00007	0.00018	0.00289	0.0001	0.00023
		By Fuel	0.00007	0.00018	0.00292	0.0001	0.00023
		All to SRS	0.00007	0.00018	0.00292	0.0001	0.00023
	NTS/ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00048
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00022	0.00052	0.00245	0.0022	0.00097
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024
	HS/SRS	Geographic	0.00007	0.00018	0.00289	0.0001	0.00023
		By Fuel	0.00007	0.00018	0.00292	0.0001	0.00023
		All to SRS	0.00007	0.00018	0.00292	0.0001	0.00023
	HS/ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00048
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00019	0.00045	0.00219	0.0016	0.00095
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024
Centralization	INEL		0.00011	0.00017	0.00231	0.0003	0.00100
	SRS		0.00007	0.00018	0.00292	0.0001	0.00023
	HS	Geographic	0.00050	0.00150	0.00333	0.0080	0.00255
		By Fuel	0.00020	0.00028	0.00377	0.0005	0.00082
		All to INEL	0.00020	0.00046	0.00243	0.0016	0.00116
		All to SRS	0.00020	0.00028	0.00377	0.0005	0.00082
	NTS	Geographic	0.00046	0.00131	0.00339	0.0077	0.00251
		By Fuel	0.00019	0.00027	0.00380	0.0004	0.00079
		All to INEL	0.00023	0.00054	0.00269	0.0022	0.00117
		All to SRS	0.00019	0.00027	0.00380	0.0004	0.00079
	ORR	Geographic	0.00014	0.00036	0.00267	0.0014	0.00049
		By Fuel	0.00010	0.00019	0.00267	0.0002	0.00024
		All to INEL	0.00041	0.00116	0.00245	0.0064	0.00266
		All to SRS	0.00010	0.00019	0.00267	0.0002	0.00024

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-37 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Truck, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radio-logical
Decentralization	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
1992/1993 Planning Basis	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
Regionalization by Fuel Type	INEL/SRS		0.00024	0.00068	0.00021	0.0044	0.00080
Regionalization	INEL/SRS		0.00024	0.00066	0.00019	0.0042	0.00074
by	INEL/ORR	Geographic	0.00030	0.00084	0.00030	0.0053	0.00100
Geography		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00043	0.00131	0.00062	0.0067	0.00355
	NTS/SRS	Geographic	0.00024	0.00066	0.00019	0.0043	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	NTS/ORR	Geographic	0.00031	0.00084	0.00031	0.0054	0.00101
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00054	0.00167	0.00098	0.0086	0.00372
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107
	HS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	HS/ORR	Geographic	0.00031	0.00084	0.00031	0.0054	0.00101
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00051	0.00160	0.00072	0.0081	0.00370
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107
Centralization	INEL		0.00049	0.00152	0.00077	0.0076	0.00442
	SRS		0.00024	0.00068	0.00021	0.0044	0.00080
	HS	Geographic	0.00073	0.00224	0.00091	0.0131	0.00376
		By Fuel	0.00074	0.00229	0.00094	0.0134	0.00387
		All to INEL	0.00059	0.00187	0.00088	0.0093	0.00459
		All to SRS	0.00074	0.00229	0.00094	0.0134	0.00387
	NTS	Geographic	0.00068	0.00203	0.00098	0.0129	0.00371
		By Fuel	0.00070	0.00207	0.00101	0.0131	0.00381
		All to INEL	0.00062	0.00192	0.00116	0.0099	0.00459
		All to SRS	0.00070	0.00207	0.00101	0.0131	0.00381
	ORR	Geographic	0.00031	0.00086	0.00031	0.0055	0.00105
		By Fuel	0.00031	0.00086	0.00033	0.0055	0.00107
		All to INEL	0.00074	0.00231	0.00098	0.0129	0.00541
		All to SRS	0.00031	0.00086	0.00033	0.0055	0.00107

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-38 Tabulation of Overland Transportation Risks: Accept Target Material Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.00026	0.00067	0.00028	0.0041	0.00076
Geography		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00043	0.00131	0.00062	0.0067	0.00355
	NTS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	NTS/ORR	Geographic	0.00027	0.00067	0.00029	0.0041	0.00076
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00047	0.00133	0.00080	0.0068	0.00357
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082
	HS/SRS	Geographic	0.00024	0.00066	0.00019	0.0042	0.00074
		By Fuel	0.00024	0.00068	0.00021	0.0044	0.00080
		All to SRS	0.00024	0.00068	0.00021	0.0044	0.00080
	HS/ORR	Geographic	0.00027	0.00067	0.00029	0.0041	0.00076
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00047	0.00133	0.00080	0.0068	0.00358
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082
Centralization	INEL						
	SRS						
	HS	Geographic	0.00042	0.00101	0.00129	0.0055	0.00201
		By Fuel	0.00043	0.00104	0.00133	0.0057	0.00208
		All to INEL	0.00055	0.00160	0.00096	0.0080	0.00448
		All to SRS	0.00043	0.00104	0.00133	0.0057	0.00208
	NTS	Geographic	0.00041	0.00098	0.00134	0.0056	0.00196
		By Fuel	0.00042	0.00100	0.00138	0.0058	0.00203
		All to INEL	0.00054	0.00158	0.00098	0.0081	0.00444
		All to SRS	0.00042	0.00100	0.00138	0.0058	0.00203
	ORR	Geographic	0.00027	0.00067	0.00030	0.0041	0.00077
		By Fuel	0.00027	0.00069	0.00031	0.0043	0.00082
		All to INEL	0.00051	0.00137	0.00118	0.0069	0.00390
		All to SRS	0.00027	0.00069	0.00031	0.0043	0.00082

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-39 Tabulation of Overland Transportation Risks: Accept Target Material Only, All Shipments via Rail, Upper Bound Risk Factors, Risk Increases over that of the Basic Implementation

Alternative / Option			Routine			Accidental	
			Radiological		Nonradiological		Radio-logical
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
1992/1993 Planning Basis	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
Regionalization by Fuel Type	INEL/SRS		0.00008	0.00019	0.00305	0.0002	0.00026
Regionalization by Geography	INEL/SRS		0.00008	0.00019	0.00301	0.0002	0.00025
	INEL/ORR	Geographic	0.00010	0.00019	0.00276	0.0002	0.00025
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00010	0.00016	0.00242	0.0002	0.00089
	NTS/SRS	Geographic	0.00008	0.00019	0.00302	0.0002	0.00025
		By Fuel	0.00008	0.00019	0.00305	0.0002	0.00026
		All to SRS	0.00008	0.00019	0.00305	0.0002	0.00026
	NTS/ORR	Geographic	0.00015	0.00037	0.00278	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00022	0.00053	0.00278	0.0022	0.00106
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026
	HS/SRS	Geographic	0.00008	0.00019	0.00302	0.0002	0.00025
		By Fuel	0.00008	0.00019	0.00305	0.0002	0.00026
		All to SRS	0.00008	0.00019	0.00305	0.0002	0.00026
	HS/ORR	Geographic	0.00015	0.00037	0.00278	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00019	0.00045	0.00252	0.0016	0.00104
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026
Centralization	INEL		0.00012	0.00018	0.00268	0.0003	0.00111
	SRS		0.00008	0.00019	0.00305	0.0002	0.00026
	HS	Geographic	0.00051	0.00151	0.00352	0.0080	0.00258
		By Fuel	0.00020	0.00029	0.00396	0.0005	0.00086
		All to INEL	0.00021	0.00047	0.00282	0.0016	0.00127
		All to SRS	0.00020	0.00029	0.00396	0.0005	0.00086
	NTS	Geographic	0.00046	0.00132	0.00354	0.0077	0.00255
		By Fuel	0.00019	0.00028	0.00396	0.0005	0.00082
		All to INEL	0.00024	0.00055	0.00306	0.0022	0.00128
		All to SRS	0.00019	0.00028	0.00396	0.0005	0.00082
	ORR	Geographic	0.00015	0.00037	0.00279	0.0014	0.00050
		By Fuel	0.00011	0.00020	0.00280	0.0002	0.00026
		All to INEL	0.00041	0.00117	0.00278	0.0064	0.00275
		All to SRS	0.00011	0.00020	0.00280	0.0002	0.00026

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-40 Potential Consequences for the Most Severe Accidents Involving Shipments of Target Material[a,b]

Mode and Accident Location	Neutral Conditions[c]				Stable Conditions[d]			
	Population[e]		MEI[f]		Population[e]		MEI[f]	
	Dose (person-rem)	Consequences (LCF)	Dose (rem)	Consequences (LCF)	Dose (person-rem)	Consequences (LCF)	Dose (person-rem)	Consequences (LCF)
Truck:								
Urban	206	0.1	0.15	0.000074	1650	0.83	0.50	0.00025
Suburban	38.3	0.019	0.15	0.000074	307	0.15	0.50	0.00025
Rural	0.70	0.00035	0.15	0.000074	5.5	0.0028	0.50	0.00025
Rail:								
Urban	206	0.1	0.15	0.000074	1650	0.83	0.50	0.00025
Suburban	38.3	0.19	0.15	0.000074	307	0.15	0.50	0.00025
Rural	0.70	0.00035	0.15	0.000074	5.5	0.0028	0.50	0.00025

[a] *The most severe accidents correspond to modal study accident severity category 6 (DOE, 1994b).*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform population density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation, acute cloudshine, groundshine, resuspended inhalation, resuspended cloudshine, and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

E.8.5 Implementation Alternative - Implementing an Acceptance Policy for Varying Durations - Indefinite HEU Acceptance

Since most LEU would come back within 10 years and spent nuclear fuel produced from the indefinite operation of HEU reactors is difficult to predict, it is reasonable to assume that the analysis for the basic implementation applies closely.

E.8.6 Implementation Alternative - Implementing an Acceptance Policy with Varying Financial Approaches

None of the financial approaches would have a significant effect on overland transportation. The effects calculated for the basic implementation adequately model this strategy.

Table E-41 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
1992/1993 Planning Basis	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
Regionalization by Fuel Type	INEL/SRS		0.030	0.093	0.004	0.056	0.000032
Regionalization	INEL/SRS		0.015	0.045	0.002	0.028	0.000013
by	INEL/ORR	Geographic	0.017	0.051	0.002	0.032	0.000046
Geography		By Fuel	0.032	0.099	0.005	0.060	0.000066
		All to INEL	0.050	0.157	0.008	0.090	0.000055
	NTS/SRS	Geographic	0.016	0.049	0.002	0.030	0.000022
		By Fuel	0.031	0.096	0.005	0.058	0.000041
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	NTS/ORR	Geographic	0.018	0.055	0.003	0.034	0.000055
		By Fuel	0.033	0.102	0.005	0.062	0.000074
		All to INEL	0.054	0.171	0.009	0.097	0.000092
		All to SRS	0.030	0.090	0.005	0.055	0.000072
	HS/SRS	Geographic	0.016	0.048	0.002	0.030	0.000016
		By Fuel	0.031	0.095	0.005	0.057	0.000035
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	HS/ORR	Geographic	0.018	0.054	0.002	0.033	0.000049
		By Fuel	0.033	0.101	0.005	0.061	0.000068
		All to INEL	0.053	0.168	0.008	0.095	0.000065
		All to SRS	0.030	0.090	0.005	0.055	0.000072
Centralization	INEL		0.050	0.157	0.008	0.090	0.000055
	SRS		0.027	0.082	0.004	0.050	0.000029
	HS	Geographic	0.029	0.088	0.004	0.053	0.000133
		By Fuel	0.044	0.136	0.006	0.081	0.000153
		All to INEL	0.053	0.168	0.008	0.095	0.000065
		All to SRS	0.044	0.134	0.006	0.081	0.000183
	NTS	Geographic	0.028	0.083	0.004	0.053	0.000152
		By Fuel	0.043	0.130	0.007	0.081	0.000172
		All to INEL	0.054	0.171	0.009	0.097	0.000092
		All to SRS	0.042	0.127	0.006	0.080	0.000199
	ORR	Geographic	0.020	0.060	0.003	0.038	0.000071
		By Fuel	0.035	0.108	0.005	0.065	0.000090
		All to INEL	0.061	0.195	0.009	0.114	0.000162
		All to SRS	0.030	0.090	0.005	0.055	0.000072

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-42 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, Shipments from Ports via Truck, Intersite Shipments via Rail, Average Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.016	0.046	0.002	0.028	0.000017
		By Fuel	0.031	0.093	0.005	0.056	0.000036
		All to INEL	0.050	0.157	0.008	0.090	0.000055
	NTS/SRS	Geographic	0.015	0.046	0.002	0.028	0.000014
		By Fuel	0.030	0.093	0.004	0.056	0.000033
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	NTS/ORR	Geographic	0.016	0.046	0.002	0.028	0.000017
		By Fuel	0.031	0.093	0.005	0.056	0.000037
		All to INEL	0.050	0.157	0.008	0.090	0.000058
		All to SRS	0.028	0.083	0.004	0.050	0.000033
	HS/SRS	Geographic	0.015	0.046	0.002	0.028	0.000014
		By Fuel	0.030	0.093	0.004	0.056	0.000033
		All to SRS	0.027	0.082	0.004	0.050	0.000029
	HS/ORR	Geographic	0.016	0.046	0.002	0.028	0.000018
		By Fuel	0.031	0.093	0.005	0.056	0.000037
		All to INEL	0.050	0.157	0.008	0.090	0.000059
		All to SRS	0.028	0.083	0.004	0.050	0.000033
Centralization	INEL						
	SRS						
	HS	Geographic	0.017	0.047	0.003	0.029	0.000039
		By Fuel	0.032	0.094	0.006	0.057	0.000058
		All to INEL	0.050	0.157	0.008	0.090	0.000059
		All to SRS	0.029	0.084	0.006	0.050	0.000061
	NTS	Geographic	0.017	0.047	0.003	0.029	0.000037
		By Fuel	0.032	0.094	0.006	0.056	0.000057
		All to INEL	0.050	0.157	0.008	0.090	0.000058
		All to SRS	0.029	0.084	0.006	0.050	0.000060
	ORR	Geographic	0.016	0.046	0.002	0.028	0.000022
		By Fuel	0.031	0.093	0.005	0.056	0.000041
		All to INEL	0.051	0.158	0.008	0.090	0.000076
		All to SRS	0.028	0.083	0.004	0.050	0.000033

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-43 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Average Risk Factors

Alternative / Option			Routine				Accidental
			Radiological		Nonradiological		Radio-
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	logical
Decentralization	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
1992/1993 Planning Basis	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
Regionalization by Fuel Type	INEL/SRS		0.009	0.011	0.015	0.002	0.000010
Regionalization	INEL/SRS		0.006	0.007	0.009	0.001	0.000005
by	INEL/ORR	Geographic	0.007	0.007	0.009	0.001	0.000008
Geography		By Fuel	0.010	0.012	0.015	0.002	0.000013
		All to INEL	0.013	0.013	0.018	0.003	0.000013
	NTS/SRS	Geographic	0.006	0.007	0.009	0.001	0.000006
		By Fuel	0.009	0.011	0.015	0.002	0.000011
		All to SRS	0.009	0.011	0.014	0.002	0.000010
	NTS/ORR	Geographic	0.009	0.013	0.009	0.005	0.000039
		By Fuel	0.011	0.015	0.015	0.004	0.000022
		All to INEL	0.017	0.027	0.020	0.010	0.000050
		All to SRS	0.009	0.011	0.014	0.002	0.000014
	HS/SRS	Geographic	0.006	0.007	0.009	0.001	0.000006
		By Fuel	0.009	0.011	0.015	0.002	0.000011
		All to SRS	0.009	0.011	0.014	0.002	0.000010
	HS/ORR	Geographic	0.009	0.013	0.009	0.005	0.000039
		By Fuel	0.010	0.014	0.015	0.003	0.000016
		All to INEL	0.016	0.025	0.019	0.008	0.000023
		All to SRS	0.009	0.011	0.014	0.002	0.000014
Centralization	INEL		0.013	0.013	0.018	0.003	0.000013
	SRS		0.009	0.011	0.014	0.002	0.000010
	HS	Geographic	0.019	0.047	0.010	0.024	0.000124
		By Fuel	0.011	0.015	0.016	0.003	0.000037
		All to INEL	0.016	0.025	0.019	0.008	0.000023
		All to SRS	0.010	0.013	0.015	0.002	0.000042
	NTS	Geographic	0.018	0.041	0.010	0.024	0.000136
		By Fuel	0.012	0.016	0.016	0.004	0.000042
		All to INEL	0.017	0.027	0.020	0.010	0.000050
		All to SRS	0.010	0.012	0.015	0.002	0.000040
	ORR	Geographic	0.009	0.013	0.009	0.005	0.000043
		By Fuel	0.012	0.020	0.015	0.007	0.000038
		All to INEL	0.025	0.052	0.020	0.027	0.000120
		All to SRS	0.009	0.011	0.014	0.002	0.000014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-44 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
1992/1993 Planning Basis	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
Regionalization by Fuel Type	INEL/SRS		0.025	0.078	0.003	0.044	0.000010
Regionalization by Geography	INEL/SRS		0.011	0.033	0.001	0.019	0.000005
	INEL/ORR	Geographic	0.013	0.039	0.001	0.023	0.000038
		By Fuel	0.027	0.084	0.003	0.048	0.000043
		All to INEL	0.041	0.131	0.006	0.073	0.000018
	NTS/SRS	Geographic	0.012	0.036	0.001	0.021	0.000014
		By Fuel	0.026	0.081	0.003	0.046	0.000019
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	NTS/ORR	Geographic	0.014	0.042	0.002	0.025	0.000047
		By Fuel	0.029	0.087	0.003	0.049	0.000052
		All to INEL	0.046	0.145	0.007	0.081	0.000055
		All to SRS	0.026	0.079	0.003	0.044	0.000053
	HS/SRS	Geographic	0.012	0.035	0.001	0.021	0.000008
		By Fuel	0.026	0.081	0.003	0.045	0.000013
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	HS/ORR	Geographic	0.014	0.041	0.001	0.025	0.000041
		By Fuel	0.028	0.087	0.003	0.049	0.000046
		All to INEL	0.045	0.143	0.006	0.078	0.000028
		All to SRS	0.026	0.079	0.003	0.044	0.000053
Centralization	INEL		0.041	0.131	0.006	0.073	0.000018
	SRS		0.023	0.071	0.002	0.039	0.000010
	HS	Geographic	0.024	0.075	0.003	0.044	0.000125
		By Fuel	0.039	0.121	0.004	0.069	0.000131
		All to INEL	0.045	0.143	0.006	0.078	0.000028
		All to SRS	0.040	0.123	0.004	0.070	0.000164
	NTS	Geographic	0.023	0.070	0.003	0.044	0.000144
		By Fuel	0.038	0.116	0.005	0.068	0.000149
		All to INEL	0.046	0.145	0.007	0.081	0.000055
		All to SRS	0.038	0.115	0.005	0.068	0.000180
	ORR	Geographic	0.016	0.048	0.002	0.029	0.000064
		By Fuel	0.030	0.093	0.003	0.053	0.000068
		All to INEL	0.053	0.170	0.007	0.097	0.000125
		All to SRS	0.026	0.079	0.003	0.044	0.000053

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-45 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments from Ports via Truck, Intersite Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine				Accidental
			Radiological		Nonradiological		Radio-
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	logical
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization	INEL/SRS						
by	INEL/ORR	Geographic	0.011	0.033	0.001	0.020	0.000009
Geography		By Fuel	0.026	0.078	0.003	0.044	0.000014
		All to INEL	0.041	0.131	0.006	0.073	0.000018
	NTS/SRS	Geographic	0.011	0.033	0.001	0.020	0.000006
		By Fuel	0.025	0.078	0.003	0.044	0.000011
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	NTS/ORR	Geographic	0.011	0.033	0.001	0.020	0.000009
		By Fuel	0.026	0.078	0.003	0.044	0.000014
		All to INEL	0.042	0.132	0.006	0.073	0.000021
		All to SRS	0.024	0.071	0.002	0.039	0.000014
	HS/SRS	Geographic	0.011	0.033	0.001	0.020	0.000006
		By Fuel	0.025	0.078	0.003	0.044	0.000011
		All to SRS	0.023	0.071	0.002	0.039	0.000010
	HS/ORR	Geographic	0.011	0.033	0.001	0.020	0.000010
		By Fuel	0.026	0.078	0.003	0.044	0.000015
		All to INEL	0.042	0.132	0.006	0.073	0.000022
		All to SRS	0.024	0.071	0.002	0.039	0.000014
Centralization	INEL						
	SRS						
	HS	Geographic	0.012	0.034	0.002	0.020	0.000031
		By Fuel	0.027	0.079	0.004	0.044	0.000036
		All to INEL	0.042	0.132	0.006	0.073	0.000022
		All to SRS	0.025	0.073	0.004	0.039	0.000042
	NTS	Geographic	0.012	0.034	0.002	0.020	0.000030
		By Fuel	0.027	0.079	0.004	0.044	0.000035
		All to INEL	0.042	0.132	0.006	0.073	0.000021
		All to SRS	0.025	0.072	0.004	0.039	0.000041
	ORR	Geographic	0.012	0.033	0.001	0.020	0.000014
		By Fuel	0.026	0.079	0.003	0.044	0.000019
		All to INEL	0.042	0.132	0.007	0.074	0.000039
		All to SRS	0.024	0.071	0.002	0.039	0.000014

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-46 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Lower Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
1992/1993 Planning Basis	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
Regionalization by Fuel Type	INEL/SRS		0.008	0.009	0.011	0.001	0.000004
Regionalization by Geography	INEL/SRS		0.005	0.005	0.006	0.001	0.000003
	INEL/ORR	Geographic	0.006	0.006	0.007	0.001	0.000006
		By Fuel	0.008	0.009	0.011	0.002	0.000007
		All to INEL	0.011	0.009	0.012	0.002	0.000003
	NTS/SRS	Geographic	0.005	0.006	0.006	0.001	0.000004
		By Fuel	0.008	0.009	0.011	0.002	0.000004
		All to SRS	0.008	0.009	0.011	0.001	0.000004
	NTS/ORR	Geographic	0.007	0.011	0.007	0.004	0.000037
		By Fuel	0.009	0.012	0.012	0.003	0.000016
		All to INEL	0.015	0.023	0.013	0.010	0.000040
		All to SRS	0.008	0.009	0.011	0.001	0.000008
	HS/SRS	Geographic	0.005	0.006	0.006	0.001	0.000004
		By Fuel	0.008	0.009	0.011	0.002	0.000005
		All to SRS	0.008	0.009	0.011	0.001	0.000004
	HS/ORR	Geographic	0.007	0.011	0.007	0.004	0.000037
		By Fuel	0.009	0.012	0.011	0.003	0.000009
		All to INEL	0.014	0.020	0.012	0.008	0.000013
		All to SRS	0.008	0.009	0.011	0.001	0.000008
Centralization	INEL		0.011	0.009	0.012	0.002	0.000003
	SRS		0.008	0.009	0.011	0.001	0.000004
	HS	Geographic	0.018	0.045	0.008	0.024	0.000122
		By Fuel	0.010	0.013	0.012	0.003	0.000031
		All to INEL	0.014	0.020	0.012	0.008	0.000013
		All to SRS	0.009	0.011	0.012	0.002	0.000036
	NTS	Geographic	0.017	0.040	0.008	0.023	0.000133
		By Fuel	0.010	0.013	0.013	0.004	0.000036
		All to INEL	0.015	0.023	0.013	0.010	0.000040
		All to SRS	0.009	0.010	0.013	0.002	0.000035
	ORR	Geographic	0.008	0.011	0.007	0.004	0.000041
		By Fuel	0.011	0.018	0.012	0.007	0.000032
		All to INEL	0.023	0.048	0.013	0.026	0.000110
		All to SRS	0.008	0.009	0.011	0.001	0.000008

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-47 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Truck, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
			Radiological		Nonradiological		Radio-logical
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
1992/1993 Planning Basis	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
Regionalization by Fuel Type	INEL/SRS		0.039	0.118	0.008	0.073	0.00009
Regionalization by Geography	INEL/SRS		0.027	0.078	0.005	0.046	0.00006
	INEL/ORR	Geographic	0.029	0.083	0.006	0.050	0.00009
		By Fuel	0.041	0.124	0.009	0.077	0.00012
		All to INEL	0.052	0.165	0.010	0.101	0.00011
	NTS/SRS	Geographic	0.028	0.081	0.006	0.048	0.00006
		By Fuel	0.040	0.121	0.009	0.075	0.00010
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	NTS/ORR	Geographic	0.030	0.087	0.006	0.052	0.00010
		By Fuel	0.042	0.127	0.009	0.078	0.00013
		All to INEL	0.057	0.179	0.011	0.109	0.00015
		All to SRS	0.041	0.120	0.009	0.073	0.00013
	HS/SRS	Geographic	0.027	0.080	0.005	0.047	0.00006
		By Fuel	0.040	0.121	0.008	0.074	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	HS/ORR	Geographic	0.030	0.086	0.006	0.051	0.00009
		By Fuel	0.042	0.126	0.009	0.078	0.00013
		All to INEL	0.056	0.176	0.010	0.107	0.00012
		All to SRS	0.041	0.120	0.009	0.073	0.00013
Centralization	INEL		0.052	0.165	0.010	0.101	0.00011
	SRS		0.038	0.113	0.008	0.068	0.00009
	HS	Geographic	0.040	0.120	0.007	0.071	0.00018
		By Fuel	0.053	0.161	0.010	0.098	0.00021
		All to INEL	0.056	0.176	0.010	0.107	0.00012
		All to SRS	0.054	0.165	0.011	0.099	0.00024
	NTS	Geographic	0.039	0.115	0.007	0.071	0.00019
		By Fuel	0.052	0.156	0.010	0.097	0.00023
		All to INEL	0.057	0.179	0.011	0.109	0.00015
		All to SRS	0.053	0.157	0.011	0.098	0.00026
	ORR	Geographic	0.032	0.093	0.006	0.056	0.00011
		By Fuel	0.044	0.133	0.009	0.082	0.00015
		All to INEL	0.064	0.203	0.011	0.125	0.00022
		All to SRS	0.041	0.120	0.009	0.073	0.00013

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-48 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments from Ports via Truck, Intersite Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological		Nonradiological		Radio-logical
			Crew	Public	Emis.	Traffic	
Decentralization	INEL/SRS						
1992/1993 Planning Basis	INEL/SRS						
Regionalization by Fuel Type	INEL/SRS						
Regionalization by Geography	INEL/SRS						
	INEL/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.052	0.165	0.010	0.101	0.00011
	NTS/SRS	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.039	0.118	0.008	0.073	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	NTS/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.038	0.113	0.009	0.069	0.00009
	HS/SRS	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.039	0.118	0.008	0.073	0.00009
		All to SRS	0.038	0.113	0.008	0.068	0.00009
	HS/ORR	Geographic	0.027	0.078	0.005	0.046	0.00006
		By Fuel	0.040	0.118	0.008	0.073	0.00009
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.038	0.113	0.009	0.069	0.00009
Centralization	INEL						
	SRS						
	HS	Geographic	0.028	0.079	0.006	0.047	0.00008
		By Fuel	0.041	0.119	0.009	0.073	0.00012
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.040	0.114	0.010	0.069	0.00012
	NTS	Geographic	0.028	0.079	0.006	0.047	0.00008
		By Fuel	0.041	0.119	0.009	0.073	0.00011
		All to INEL	0.053	0.165	0.010	0.102	0.00012
		All to SRS	0.040	0.114	0.010	0.069	0.00012
	ORR	Geographic	0.027	0.078	0.006	0.046	0.00006
		By Fuel	0.040	0.118	0.009	0.073	0.00010
		All to INEL	0.054	0.165	0.011	0.102	0.00014
		All to SRS	0.038	0.113	0.009	0.069	0.00009

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

Table E-49 Tabulation of Overland Transportation Risks: Five-Year Spent Nuclear Fuel Acceptance Only, All Shipments via Rail, Upper Bound Risk Factors

Alternative / Option			Routine			Accidental	
Programmatic SNF & INEL EIS Alternative	SNF Site Option	Phase I Approach	Radiological Crew	Radiological Public	Nonradiological Emis.	Nonradiological Traffic	Radiological
Decentralization	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
1992/1993 Planning Basis	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
Regionalization by Fuel Type	INEL/SRS		0.011	0.016	0.026	0.002	0.00003
Regionalization by Geography	INEL/SRS		0.008	0.012	0.016	0.002	0.00002
	INEL/ORR	Geographic	0.009	0.013	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00004
		All to INEL	0.013	0.016	0.039	0.003	0.00004
	NTS/SRS	Geographic	0.009	0.012	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00003
		All to SRS	0.011	0.017	0.022	0.002	0.00003
	NTS/ORR	Geographic	0.011	0.018	0.016	0.005	0.00005
		By Fuel	0.012	0.020	0.026	0.004	0.00004
		All to INEL	0.018	0.030	0.041	0.011	0.00008
		All to SRS	0.011	0.017	0.022	0.002	0.00004
	HS/SRS	Geographic	0.008	0.012	0.016	0.002	0.00002
		By Fuel	0.011	0.016	0.026	0.002	0.00003
		All to SRS	0.011	0.017	0.022	0.002	0.00003
	HS/ORR	Geographic	0.011	0.018	0.016	0.005	0.00005
		By Fuel	0.012	0.019	0.026	0.004	0.00004
		All to INEL	0.017	0.027	0.040	0.008	0.00005
		All to SRS	0.011	0.017	0.022	0.002	0.00004
Centralization	INEL		0.013	0.016	0.039	0.003	0.00004
	SRS		0.011	0.017	0.022	0.002	0.00003
	HS	Geographic	0.021	0.052	0.017	0.025	0.00014
		By Fuel	0.013	0.020	0.027	0.004	0.00006
		All to INEL	0.017	0.027	0.040	0.008	0.00005
		All to SRS	0.012	0.018	0.023	0.003	0.00006
	NTS	Geographic	0.020	0.046	0.017	0.024	0.00015
		By Fuel	0.013	0.020	0.027	0.004	0.00007
		All to INEL	0.018	0.030	0.041	0.011	0.00008
		All to SRS	0.012	0.018	0.023	0.003	0.00006
	ORR	Geographic	0.011	0.018	0.016	0.005	0.00006
		By Fuel	0.014	0.025	0.026	0.008	0.00006
		All to INEL	0.025	0.055	0.041	0.027	0.00015
		All to SRS	0.011	0.017	0.022	0.002	0.00004

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

E.8.7 Implementation Alternative - Implementing an Acceptance Policy by Taking Title at Varying Locations

The agency that has title to the spent nuclear fuel has no significant effect on overland transportation. The effects calculated for the basic implementation apply here.

E.8.8 Implementation Alternative - Implementing an Acceptance Policy and Storing Underwater

The use of underwater storage would have only minor effects on the location to which foreign research reactor spent nuclear fuel were delivered on the DOE sites. However, since there is some degree of uncertainty in the exact delivery location on all the DOE sites and intrasite transportation would be less likely, the effects calculated for the basic implementation apply here.

E.8.9 Implementation Alternative - Implementing an Acceptance Policy and Near-Term Chemical Separation in the United States

The performance of conventional or alternative chemical separation is only considered feasible at the Idaho National Engineering Laboratory and Savannah River Site sites. The requirements for overland transportation are not affected by the activities at the sites. Therefore, the impacts calculated in Section E.7 for the options to transport fuel to Idaho National Engineering Laboratory and/or Savannah River Site under the Regionalization by Fuel Type or Centralization alternatives would apply to this section. They are shown in Table E-50.

Table E-50 Tabulation of Overland Transportation Risks: Chemical Separation in the United States

Alternative/Option			*Incident-free*				*Accidental*
			Radiological		*Nonradiological*		
Implementation	*Mode*	*Risk Factors*	*Crew*	*Public*	*Emis.*	*Traffic*	*Radiological*
Regionalization by Fuel Type	Truck	Upper	0.048	0.143	0.010	0.088	0.000109
		Nominal	0.036	0.112	0.005	0.067	0.000039
		Lower	0.030	0.093	0.003	0.052	0.000012
	Rail	Upper	0.013	0.020	0.031	0.003	0.000040
		Nominal	0.011	0.014	0.018	0.002	0.000012
		Lower	0.010	0.011	0.014	0.002	0.000005
Centralization to Idaho National Engineering Laboratory	Truck	Upper	0.065	0.205	0.012	0.126	0.000143
		Nominal	0.062	0.195	0.009	0.112	0.000069
		Lower	0.051	0.163	0.007	0.091	0.000023
	Rail	Upper	0.016	0.020	0.049	0.004	0.000053
		Nominal	0.016	0.016	0.023	0.004	0.000016
		Lower	0.013	0.011	0.015	0.003	0.000004
Centralization to Savannah River Site	Truck	Upper	0.046	0.137	0.010	0.083	0.000107
		Nominal	0.033	0.097	0.005	0.062	0.000035
		Lower	0.028	0.085	0.003	0.047	0.000012
	Rail	Upper	0.013	0.020	0.027	0.003	0.000038
		Nominal	0.011	0.013	0.017	0.002	0.000012
		Lower	0.009	0.011	0.013	0.002	0.000005

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel policy, except for the Accidental - Traffic column, which is the number of fatalities during the policy.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.020 to 0.27 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and, 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.009 to 0.065. The estimated number of radiation-related LCFs for the general population ranged from 0.011 to 0.21, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.05.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 0.000004 to 0.00014 LCFs from radiation and from 0.002 to 0.13 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Table E-50. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

The consequences of the most severe accident hypothesized are the same as described for the Basic Implementation since the material at risk is the same.

E.8.10 Developmental Processing Capabilities

The overland transportation impacts would be based on the site selected for processing, and would be determined after that site is selected.

E.8.11 Management Alternative - Adopt a Strategy of Managing Foreign Research Reactor Spent Nuclear Fuel Overseas: Store Overseas

There would be no overland transportation impacts in the United States if this alternative were implemented.

E.8.12 Policy Alternative - Adopt a Strategy of Managing Foreign Research Reactor Spent Nuclear Fuel Overseas: Process Overseas and Ship Vitrified High-Level Waste to the United States

The total amount of foreign research reactor spent nuclear fuel could be reduced into 16 vitrified waste logs, which could be carried in 8 casks. The contents of each cask is described isotopically in Table E-3. The curie content is based on the total number of curies expected to be returned to the United States under the basic implementation of Management Alternative 1. Realistically, the logs might have to be allowed to decay at the vitrification facility until the dose rate was below the regulatory-limit. Therefore, all incident-free calculations assume the dose rate is 10 mrem per hr at 2 m (6.6 ft).

This alternative is assumed to be independent of the SNF&INEL Final EIS (DOE, 1995) results. The only site considered for interim storage of vitrified high-level waste is the Savannah River Site. The only overseas facilities are in Europe, so all shipments are assumed to arrive on the east coast.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.0002 to 0.004 over the entire duration of the program. These results are the sum of the estimated number of radiation-related LCFs to the public and the crews.

The range of fatality estimates is caused by the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.00014 to 0.001. The estimated number of radiation-related LCFs for the general population ranged from 0.00009 to 0.003, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.0001 to 0.0005.

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 1.9×10^{-7} to 5.9×10^{-6} LCF from radiation and from 0.00003 to 0.002 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Tables E-51 and E-52.

Table E-51 Tabulation of Ground Transportation Risks: Vitrified High-Level Waste Acceptance Only

Alternative/Option			*Incident-free*				*Accidental*
			Radiological		*Nonradiological*		
Implementation	*Mode*	*Risk Factors*	*Crew*	*Public*	*Emis.*	*Traffic*	*Radiological*
Ship directly to repository	Truck	Upper	0.00076	0.00240	0.00016	0.00162	5.9×10^{-6}
		Nominal	0.00072	0.00227	0.00013	0.00143	5.0×10^{-6}
		Lower	0.00053	0.00172	0.00010	0.00106	1.7×10^{-6}
	Rail	Upper	0.00020	0.00024	0.00052	0.00005	2.0×10^{-6}
		Nominal	0.00019	0.00019	0.00025	0.00005	1.2×10^{-6}
		Lower	0.00014	0.00009	0.00015	0.00003	1.9×10^{-7}
Ship to Savannah River Site, then to repository	Truck	Upper	0.00102	0.00302	0.00018	0.00196	1.0×10^{-5}
		Nominal	0.00083	0.00249	0.00013	0.00168	7.6×10^{-6}
		Lower	0.00076	0.00227	0.00011	0.00153	6.3×10^{-6}
	Rail	Upper	0.00029	0.00030	0.00035	0.00007	2.3×10^{-6}
		Nominal	0.00025	0.00021	0.00022	0.00006	1.3×10^{-6}
		Lower	0.00023	0.00018	0.00019	0.00005	9.7×10^{-7}

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel policy, except for the Accidental - Traffic column, which is the number of fatalities during the policy.

E.8.13 Management Alternative - The Hybrid Alternative

The hybrid alternative is based on the SNF&INEL Final EIS (DOE, 1995) Regionalization by Fuel Type. The origin of shipment count is described in detail in Chapter 2 and Appendix B. The shipment count is:

	East Coast		*West Coast*		*Totals*
	Aluminum	*TRIGA*	*Aluminum*	*TRIGA*	
Phase 1	212	82	101	42	437
Phase 2	63	25	30	13	131
Total	**275**	**107**	**131**	**55**	**568**

Table E-52 Potential Consequences for the Most Severe Accidents Involving Shipments of Foreign Research Reactor High-Level Waste[a,b]

	Neutral Conditions[c]				Stable Conditions[d]			
	Population[e]		MEI[f]		Population[e]		MEI[f]	
Mode and Accident Location	Dose (person-rem)	Consequences (LCF)	Dose (rem)	Consequences (LCF)	Dose (person-rem)	Consequences (LCF)	Dose (rem)	Consequences (LCF)
Truck								
Urban	121	0.06	0.09	0.000044	970	0.48	0.29	0.00015
Suburban	22.5	0.01	0.09	0.000044	180	0.09	0.29	0.00015
Rural	0.4	0.0002	0.09	0.000044	3.2	0.002	0.29	0.00015
Rail								
Urban	121	0.06	0.09	0.000044	907	0.48	0.29	0.00015
Suburban	22.5	0.01	0.09	0.000044	180	0.09	0.29	0.00015
Rural	0.4	0.0002	0.09	0.000044	3.2	0.002	0.29	0.00015

[a] *The most severe accidents correspond to the highest NUREG-0170 accident severity category (category VIII) (NRC, 1977a). It was assumed that 0.000001 of the radioactive material would be released from its packaging and 5 percent of the aerosolized release would be respirable following an accident.*

[b] *Buoyant plume rise resulting from fire for a severe accident was included in the exposure model.*

[c] *Neutral weather conditions result in moderate dispersion and dilution of the release plume. Neutral conditions were taken to be Pasquill stability Class D with a wind speed of 4 m per sec (9 mph). Neutral conditions occur approximately 50 percent of the time in the United States.*

[d] *Stable weather conditions result in minimal dispersion and dilution of the release plume and are thus unfavorable. Stable conditions were taken to be Pasquill stability Class F with a wind speed of 1 m per sec (2.2 mph). Stable conditions occur approximately one-third of the time in the United States.*

[e] *Populations extend at a uniform density to a radius of 80 km (50 mi) from the accident site. Population exposure pathways include acute inhalation; acute cloudshine; groundshine; resuspended inhalation; resuspended cloudshine; and ingestion of food, including initially contaminated food (rural only). No decontamination or mitigative actions are taken.*

[f] *The MEI is assumed to be at the location of maximum exposure. The locations of maximum exposure would be 160 m (528 ft) and 400 m (1,320 ft) from the accident site under neutral and stable atmospheric conditions, respectively. Individual exposure pathways include acute inhalation, acute cloudshine, and groundshine during passage of the plume. No ingested dose is considered.*

No intersite shipment is necessary for this alternative. The risk estimates are summarized in Table E-53.

Impacts of Incident-Free Ground Transport

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.009 to 0.15 over the entire duration of the program. These fatalities are the sum of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates is caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and, 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.008 to 0.037. The estimated number of radiation-related LCFs for the general population ranged from 0.01 to 0.11, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.003 to 0.025.

Table E-53 Tabulation of Overland Transportation Risks: Management Alternative 3 (Hybrid Alternative)

Alternative/Option			Incident-free				Accidental	
			Radiological		Nonradiological			
Implementation	Mode	Risk Factors	Crew	Public	Emis.	Traffic	Radiological
Regionalization by Fuel Type	Truck	Upper	0.037	0.112	0.008	0.069	0.000081
		Nominal	0.033	0.098	0.005	0.058	0.000035
		Lower	0.028	0.087	0.003	0.048	0.000012
	Rail	Upper	0.010	0.015	0.025	0.002	0.000030
		Nominal	0.009	0.012	0.016	0.002	0.000011
		Lower	0.008	0.010	0.013	0.002	0.000005

All risks are expressed in latent cancer fatalities during the foreign research reactor spent nuclear fuel except for the Accidental - Traffic column, which is the number of fatalities during the policy

Impacts of Accidents During Ground Transport

The cumulative transportation accident risks over the entire program are estimated to range from 4.5 x 10^{-6} to 0.000081 LCFs from radiation and from 0.0017 to 0.069 for traffic fatality, depending on the transportation mode and DOE sites selected. Both indicate an expectation of less than one fatality.

The impacts of overland transportation are shown in Table E-53. The analysis for this implementation alternative is analogous to the analysis performed for the basic implementation of Management Alternative 1 (see Section E.7.2), and the interpretation of the tables is the same as described in Section E.7.2.

E.8.14 Transportation Implementation Example - Ship All Foreign Research Reactor Spent Nuclear Fuel to a Single Port, Regionalization-By-Fuel-Type

All the implementation alternatives analyzed in Section E.8 have been analyzed under the assumption that all foreign research reactor spent nuclear fuel would be delivered to ports on the coast nearest to the foreign research reactor (Section E.3.3). This assumption is a reasonable approximation and simplification to a complex set of possible implementation approaches. The following section, however, presents the results of the analysis associated with overland transportation risk of transporting the foreign research reactor spent nuclear fuel from a single commercial or military port.

DOE could bring all spent nuclear fuel through any identified military or commercial port. This would result in 721 shipments to that single port over the 13-year duration of the policy. Canadian fuel would be shipped overland as previously analyzed. The impacts can be directly compared with the impacts of the basic implementation of Management Alternative 1 previously analyzed under the assumption that the foreign research reactor spent nuclear fuel would arrive at the coast nearest the foreign research reactor.

Impacts of Incident-Free Ground Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in total latent fatalities that ranged from 0.017 to 0.272 over the entire duration of the program. These fatalities are the sums of the estimated number of radiation-related LCFs to the public and the crew.

The range of fatality estimates are caused by two factors: 1) the option of using truck or rail to transport spent nuclear fuel, and 2) the difference between the risk factors for the port-to-site routes.

The estimated number of radiation-related LCFs for transportation workers ranged from 0.008 to 0.069. The estimated number of radiation-related LCFs for the general population ranged from 0.009 to 0.213, and the estimated number of nonradiological fatalities from vehicular emissions ranged from 0.002 to 0.035.

Impacts of Accidents During Ground Transportation

The cumulative transportation accident risks over the entire program are estimated to range from 0.00001 to 0.00015 LCFs from radiation and from 0.001 to 0.127 for traffic fatality, depending on the transportation mode and management site(s) selected. The reasons for the range of fatality estimates are the same as those described for incident-free transportation. Both show an expectation of less than one fatality.

The consequences of the most severe accident hypothesized are the same as described for the basic implementation of Management Alternative 1 since the material at risk is the same.

Conclusion

The overland transportation risk associated with bringing all foreign research reactor spent nuclear fuel to a single port is generally within the bounds of the previous analysis that assumed that spent nuclear fuel would arrive at the coast nearest to the foreign source. In the specific case of Regionalization-By-Fuel-Type, the overland transportation risks are reduced by shipping the aluminum-based spent nuclear fuel to an east coast port. The estimated impacts of overland transportation are driven by DOE's selection of a port and the transportation mode. The increased cost and risk associated with shipping the spent nuclear fuel from Asia and Australia to the U.S. east coast are analyzed in Appendices C and F. Table E-54 gives the risk estimates associated with implementing the entire policy from each of the selected ports. The risk estimates are tabulated in a form that can be compared with the other policy and implementation alternatives analyzed in this appendix.

E.8.15 Transportation Implementation Example - Transportation by Barge

As an alternative to truck or rail transport of foreign research reactor spent nuclear fuel, barge transport from Savannah, GA to the Savannah River Site and from Portland, OR to the Hanford Site was evaluated. This section summarizes the impacts.

The analysis of the impacts of barge transportation closely parallels the analysis of truck and rail transportation described in preceding sections. Routing data was generated using the INTERLINE code for the barge routes and the HIGHWAY code for the short trucking segment. A conservative dose rate of 10 mrem per hr at 2 m (6.6 feet) from the vehicle, which is the regulatory limit, was used for calculational purposes. The RADTRAN 4 code was used to calculate the incident-free doses to the public and barge crew. The analysis of port worker consequences on breakbulk vessels was used to estimate the dose to handlers.

The barge analysis used the same radionuclide inventories used in previous sections. The RADTRAN 4 code was used to calculate the impacts of hypothetical accidental releases to the air. A specific waterborne analysis, was performed for barge accidents. Two very conservative assumptions were used in estimating the quantity of material released following an accident:

- Release fractions that determine the source term for dispersion into the water are the same as those used for similar airborne release scenarios, and

Table E-54 Tabulation of Overland Transportation Risks: Basic Implementation, All Shipments to Any Single Port, Regionalization by Fuel Type

Port	Mode	Routine				Accidental
		Radiological		Non-Radiological		
		Crew	Public	Emission	Traffic	Radiological
Charleston, SC (NWS)	Truck	0.023	0.070	0.002	0.047	0.00004
	Rail	0.008	0.009	0.011	0.001	0.00001
Charleston, SC (Wando Terminal)	Truck	0.024	0.071	0.003	0.048	0.00005
	Rail	0.008	0.009	0.011	0.001	0.00001
Galveston, TX	Truck	0.039	0.114	0.007	0.070	0.00008
	Rail	0.011	0.016	0.031	0.002	0.00003
Newport News, VA	Truck	0.031	0.093	0.005	0.060	0.00006
	Rail	0.010	0.012	0.015	0.002	0.00002
Norfolk, VA	Truck	0.031	0.091	0.003	0.060	0.00006
	Rail	0.010	0.012	0.014	0.002	0.00002
Portsmouth, VA	Truck	0.031	0.092	0.004	0.060	0.00006
	Rail	0.010	0.011	0.013	0.002	0.00002
Jacksonville, FL	Truck	0.027	0.082	0.002	0.053	0.00005
	Rail	0.009	0.009	0.011	0.001	0.00001
MOTSU, NC	Truck	0.023	0.073	0.002	0.047	0.00004
	Rail	0.009	0.010	0.011	0.002	0.00001
NWS-Concord, CA	Truck	0.069	0.213	0.019	0.127	0.00012
	Rail	0.018	0.026	0.035	0.004	0.00004
Portland, OR	Truck	0.066	0.209	0.010	0.120	0.00015
	Rail	0.018	0.021	0.027	0.004	0.00005
Savannah, GA	Truck	0.024	0.073	0.002	0.047	0.00005
	Rail	0.008	0.009	0.010	0.001	0.00001
Tacoma, WA	Truck	0.067	0.212	0.008	0.104	0.00015
	Rail	0.018	0.024	0.031	0.004	0.00005
Wilmington, NC	Truck	0.026	0.079	0.002	0.055	0.00005
	Rail	0.009	0.010	0.011	0.002	0.00001

- All of the source term resulting from an accident event is dispersed into the waterway, uniformly, over a one month period (i.e., it takes one month to recover the cask).

Barge accident statistics (Hutchinson, 1986) were used to estimate the probabilities of the accident severity classes defined in the Modal Study (Fischer et al., 1987). Barge transportation fatality statistics from Saricks and Kvitek, 1994 were used to estimate accident fatality rates. The following exposure pathways were assessed using the methodology developed by the NRC in Regulatory Guide 1.109 (NRC, 1977b):

- drinking water
- ingestion of fish
- ingestion of irrigated foods
- ingestion of meat and milk from exposed cattle
- shoreline deposits (external exposure)
- swimming (external exposure).

Collective doses were calculated for average densities for rural, suburban and urban populations, using route-specific river parameters. Additionally, MEI doses were calculated for each accident scenario in a manner analogous to that in preceding sections.

Unlike previous sections, where impacts were reported in terms of implementation of the foreign research reactor spent nuclear fuel policy, impacts are reported on a per shipment basis. As shown in Figures E-1 through E-12, the policy can be carried out in many ways, depending on the outcome of the SNF&INEL EIS (DOE, 1995) and its Record of Decision. The SNF&INEL EIS alternative that could be implemented using only barge transportation is Centralization to the Savannah River Site or the Hanford Site. All others would require various mixtures of barge transportation and overland transportation via truck or rail. Therefore, barge transportation impacts are reported on a per shipment basis and compared on that basis to shipments via truck or rail for the same origin/destination pair.

The results of the barge transportation analysis, along with comparable results from the analysis of truck and rail transportation are summarized in Table E-55.

Table E-55 Tabulation of Inland Transportation Risk Factors: Basic Implementation, Shipments via Barge to Hanford and Savannah River Sites

Alternative/ Option		*Incident Free*					*Accidental*	
		Radiological			*Nonradiological*		*Radiological*	
Port - Site	*Mode*	*Crew*	*Handlers*	*Public*	*Emission*[a]	*Traffic*	*Air-borne*	*Water-borne*
Savannah, GA to Savannah River Site	Truck	8.96×10^{-6}	N/A	0.0000277	3.22×10^{-8}	0.000021	9.35×10^{-9}	N/A
	Rail	5.44×10^{-6}	N/A	2.64×10^{-6}	5.86×10^{-7}	2.38×10^{-7}	1.11×10^{-9}	N/A
	Barge	7.64×10^{-8}	9.60×10^{-7}	1.94×10^{-6}	2.39×10^{-7}	3.42×10^{-6}	5.90×10^{-10}	2.93×10^{-8}
Portland, OR to Hanford Site	Truck	9.40×10^{-6}	N/A	0.0000294	2.67×10^{-6}	0.0000104	1.49×10^{-8}	N/A
	Rail	6.28×10^{-6}	N/A	5.75×10^{-6}	4.48×10^{-6}	5.00×10^{-7}	3.75×10^{-9}	N/A
	Barge[b]	6.40×10^{-7}	1.92×10^{-6}	4.26×10^{-6}	2.44×10^{-6}	6.69×10^{-6}	1.65×10^{-8}	1.87×10^{-8}

All risks are expressed in latent cancer fatalities during the implementation of the policy, except for the Accidental-Traffic column, which is a number of fatalities.

[a] *Assumes the same emission rate as rail transportation, and two-way travel.*

[b] *Includes truck shipment from Richland, WA to Hanford Site*

E.8.15.1 Evaluation of Barge Transportation from Portland, OR to the Hanford Site

Transportation Routes

Barge transportation from the port of Portland, OR, up the Columbia River to the town of Richland, WA, followed by truck shipment to the Hanford Site was analyzed. It was assumed that the port facilities at Portland could be used to load the casks to a barge without having to transport it into areas accessed by the public. The barge could have sailed up the river to either Richland, Pasco or Kennewick, WA. The difference in the risk parameters would be less than 5 percent. Richland was chosen for analysis because it is the largest of the three cities.

Incident-Free Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in 1.17×10^{-5} total latent fatalities per shipment. These fatalities are the sum of the estimated number of radiation-related and emission-related latent fatalities for the crew, handlers, and public.

The estimated number of radiation-related LCFs for the barge and truck crews is $6.40x10^{-7}$. The number of radiation-related LCFs for handlers during handling activities (other than the initial off-load from the seagoing ship and the on-site handling) is $1.92x10^{-6}$ per shipment. The number of radiation-related LCFs for the general population is $4.26x10^{-6}$ per shipment. The number of nonradiological fatalities from vehicle emissions is $4.88x10^{-6}$ per shipment.

The MEI risk would be the same as that in the basic implementation of Management Alternative 1, which is 0.00052 LCF for the duration of the program. This estimate is based on the conservative assumption that one individual is involved in enough driving, handling and/or inspection to reach the regulatory limit of 100 mrem per year every year for the 13-year duration of the program.

Transportation Accidents

The barge transportation accident risks from radiation exposure are estimated to be $3.63x10^{-8}$ LCF per shipment. These fatalities are the sum of the estimated number of radiation-related fatalities from atmospheric and waterborne releases. The estimated number of radiation-related LCFs from atmospheric releases is $1.65x10^{-8}$ per shipment, and $1.87x10^{-8}$ per shipment for waterborne releases. The barge transportation accident risks from other accidents than radiation are estimated to be $6.6x10^{-6}$ fatalities per shipment.

The consequences of the maximum foreseeable offsite transportation accident are 0.0295 LCF. The likelihood of this accident is approximately $1x10^{-7}$.

E.8.15.2 Evaluation of Barge Transportation from Savannah, GA to the Savannah River Site

Transportation Routes

Barge transportation from the port of Savannah, GA, up the Savannah River to the Savannah River Site was analyzed. The Savannah River Site has a barge receiving facility that could be used to off-load the casks. Handling at that facility and the onsite movement to the Receiving Basin for Offsite Fuels would not result in a significant change in calculated onsite risks (Appendix D).

Incident-Free Transportation

The incident-free transportation of spent nuclear fuel was estimated to result in $3.45x10^{-6}$ total latent fatalities per shipment. These fatalities are the sum of the estimated number of radiation-related and emission-related latent fatalities for the crew, handlers, and public.

The estimated number of radiation-related LCFs for the barge and truck crews is $7.64x10^{-8}$. The number of radiation-related LCFs for handlers during handling activities (other than the initial off-load from the seagoing ship and the on-site handling) is $9.60x10^{-7}$ per shipment. The number of radiation-related LCFs for the general population is $1.94x10^{-6}$ per shipment. The number of nonradiological fatalities from vehicle emissions is $4.97x10^{-7}$ per shipment.

The MEI risk would be the same as that in the basic implementation of Management Alternative 1, which is 0.00052 LCF for the duration of the program. This estimate is based on the conservative assumption that one individual is involved in enough driving, handling and/or inspection to reach the regulatory limit of 100 mrem per year every year for the 13-year duration of the program.

Transportation Accidents

The barge transportation accident risks are estimated to be 2.12×10^{-8} LCF per shipment. These fatalities are the sum of the estimated number of radiation-related fatalities from atmospheric and waterborne releases. The estimated number of radiation-related LCFs from atmospheric releases is 5.90×10^{-10} per shipment, and 2.93×10^{-8} per shipment for waterborne releases. The barge transportation accident risks from other than radiation are estimated to be 3.42×10^{-6} fatalities per shipment.

The consequences of the maximum foreseeable offsite transportation accident are 0.0259 LCF. The likelihood of this accident is approximately 1×10^{-7}.

E.8.15.3 Conclusions on Barge Transportation

Table E-55 provides a comparison of barge shipment parameters to truck and rail shipment parameters between the same two points. For incident-free transportation, the risk to the public and onboard crew is lower than for truck or rail shipment. However, the risk increase associated with additional handling of casks negates this risk reduction. The net incident-free risk for barge transportation is essentially identical to that for rail transportation. The radiological accident risk associated with barge transportation is larger than that of truck or rail because of the consequences of a hypothetical accident in which a damaged cask is dropped into a river. As evident from Table E-55, fatality rates for barge transportation accidents are higher than traffic fatality rates for rail shipment and lower than those for highway shipment. In total, the difference between the risks of shipping by truck, rail or barge is very low.

When traveling along a river, a barge can be observed from a long distance, and due to its slow speed, can be boarded while underway. Although not considered in detail, these characteristics would increase the vulnerability to terrorist attack.

E.9 Historical Account of Spent Nuclear Fuel Shipments and Cumulative Impacts of Transportation

E.9.1 Spent Nuclear Fuel Shipment History

The SNF&INEL Final EIS (DOE, 1995) contains a survey of transportation incidents from 1949 to 1993. For 1949 through 1970, there were 14 incidents involving irradiated fuel elements. No packages approximating a Type B shipping cask were breached as a result of these incidents. Between 1971 and 1993, there were seven transportation accidents involving spent nuclear fuel. Three involved rail shipments, and four of these accidents involved truck shipments. None of these accidents resulted in damage to the structural integrity of a cask or release of contents.

The number of spent nuclear fuel shipments and amount of spent nuclear fuel shipped throughout the entire history of spent nuclear fuel shipment cannot be precisely determined from available information. The NRC keeps accurate records of more recent (since 1979) shipments of spent nuclear fuel.

Tables E-56 and E-57 describe the spent nuclear fuel shipments in the United States that have occurred since 1979. The data for the tables comes from NUREG-0725 (NRC, 1993). These tables show detailed spent nuclear fuel shipment information, including mode of shipment (highway or rail) and shipment trends over time. Data for shipment miles are taken primarily from a road atlas and have been rounded to the nearest hundred miles for each year. Data on quantity of spent fuel shipped were provided by shippers, and have been rounded to the nearest hundred kg (220 lb) (when more than 100 kg (220 lb) were shipped). These tables do not include DOE shipments (including Naval) of spent nuclear fuel, since these shipments are not regulated by the NRC.

Table E-56 Domestic and International Spent Nuclear Fuel Shipments: 1979-1992

Year	Domestic		International		Transient
	Highway	Railway	Export	Import	
1979	2	11	0	14	0
1980	73	5	2	55	0
1981	30	2	3	48	0
1982	80	0	1	43	0
1983	92	0	2	23	0
1984	209	3	2	34	0
1985	114	18	0	21	0
1986	88	15	0	17	0
1987	85	15	3	19	0
1988	10	7	0	15	0
1989	11	6	1	4	0
1990	0	8	2	0	3
1991	7	10	4	0	1
1992	17	6	0	0	0
Totals	**818**	**106**	**20**	**293**	**4**

Source: NRC, 1993

Table E-56 shows the pattern of highway and rail shipments throughout the period 1979 to 1992. The number of shipments generally rose in the early 1980s and then declined steadily through 1992. Import shipments have generally declined since 1980, with no shipments since 1989.

Table E-57 shows that most (91.4 percent) of approximately 1,200 spent nuclear fuel shipments during the 1979 to 1992 period were completed over highways. The highway shipments accounted for a larger percentage of the mileage (94.8 percent), meaning that the longer distance shipments tended to use the highways rather than rail. However, rail shipments moved 70 percent (by weight) of the fuel. This indicates that rail has been chosen for the larger shipments over shorter distances. A review of spent nuclear fuel shipments indicates that rail transportation was often used for shipments of commercial spent nuclear fuel, and research reactors almost exclusively used trucks (NRC, 1993).

E.9.2 Cumulative Impacts of Transportation

The SNF&INEL Final EIS (DOE, 1995) analyzed the cumulative impacts of transportation, taking into account impacts from: 1) historical shipments of spent nuclear fuel to Hanford Site, Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, and Nevada Test Site; 2) the programmatic alternatives; 3) other reasonably foreseeable actions that include transportation of radioactive material; and 4) general radioactive materials transportation that is not related to a particular action.

The total worker and general population collective doses are summarized in Table E-58. Total collective worker doses from all types of shipments (historical, the alternatives, reasonably foreseeable actions, and general transportation) were estimated to be 320,000 person-rem (130 LCFs) for the period of time 1943 through 2035 (93 yr). Total general population collective doses were also estimated to be 320,000 person-rem (160 LCFs). The majority of the collective dose for workers and the general population was due to the general transportation of radioactive material. Examples of these activities are

Table E-57 Summary Data for 1979-1992 Spent Nuclear Fuel Shipment Information

Year	Number of Shipments		Kilograms Spent Fuel Shipped (thousands)[a]		Shipment Kilometers (thousands)[b]	
	Highway	Railway	Highway	Railway	Highway	Railway
1979	16	11	0.1	30.2	12.9	3.7
1980	130	5	10.0	13.6	186.6	1.6
1981	81	2	7.9	6.0	62.0	0.6
1982	124	0	7.1	0.0	171.9	0.0
1983	117	0	36.6	0.0	134.6	0.0
1984	245	3	84.5	23.8	291.9	2.6
1985	135	18	74.0	119.4	114.1	14.0
1986	105	15	40.4	97.5	77.0	14.0
1987	107	15	82.3	101.4	67.3	13.5
1988	25	7	12.8	41.8	18.4	6.9
1989	16	6	0.1	30.8	26.9	2.7
1990	2	8	0.03	65.5	2.4	1.6
1991	11	10	0.1	98.4	15.5	2.4
1992	17	6	0.1	61.3	15.7	0.8
Totals	**1,131**	**106**	**356.0**	**689.7**	**1197.2**	**64.4**

Source: NRC, 1993

[a] *To convert kilogram values to pounds, multiply values given by 2.2.*

[b] *To convert kilometer values to miles, multiply by 0.62.*

Table E-58 Cumulative Transportation-Related Radiological Collective Doses and LCFs (1943 to 2035)

Category	Collective Occupational Dose (person-rem)	Collective General Population Dose (person-rem)
Historical	200	110
Spent Nuclear Fuel Shipments for SNF&INEL Final EIS Alternatives 1-5		
Truck	1.5 to 1,000	0.34 to 2,400
Rail	1.5 to 150	0.34 to 190
Reasonably Foreseeable Actions		
Truck	11,000	50,000
Rail	820	1,700
General Transportation (1943 to 2035)	310,000	270,000
Total Collective Dose	320,000	320,000
Total LCFs	130	160

Source: DOE, 1995

shipments of radiopharmaceuticals to nuclear medicine laboratories and shipments of commercial low-level radioactive waste to commercial disposal facilities. The total number of LCFs over the time period 1943 through 2035 was estimated to be 290. Over this same period of time (93 yr), approximately 28,000,000 people would die from cancer, based on 300,000 LCFs per yr (NRC, 1977a). It should be noted that the estimated number of transportation-related LCFs would be indistinguishable from other LCFs, and the transportation-related LCFs are 0.0010 percent of the total number of LCFs.

The transportation of foreign research reactor spent nuclear fuel, under any of the proposed options or alternatives in this EIS, is included in the calculated totals under the spent nuclear fuel shipments for SNF&INEL Final EIS Alternatives 1-5 (DOE, 1995). Proposed transportation of domestic and foreign spent nuclear fuel accounts for less than one percent of the total LCFs, attributable to the transportation of radioactive material, and foreign research reactor spent nuclear fuel accounts for less than one-quarter of that one percent.

E.10 Uncertainty and Conservatism in Estimated Impacts

The sequence of analyses performed to generate the estimates of radiological risk for the transportation of spent nuclear fuel includes: 1) determination of the inventory and characteristics, 2) estimation of shipment requirements, 3) determination of route characteristics, 4) calculation of radiation doses to exposed individuals (including estimation of environmental transport and uptake of radionuclides), and 5) estimation of health effects. Uncertainties are associated with each of these steps. Uncertainties exist in the way that the physical systems being analyzed are represented by the computational models, in the data required to exercise the models (due to measurement errors, sampling errors, natural variability, or unknowns simply caused by the future nature of the actions being analyzed), and in the calculations themselves (for example, approximate algorithms used by the computers).

In principle, one can estimate the uncertainty associated with each input or computational source and predict the resultant uncertainty in each set of calculations. Thus, one can propagate the uncertainties from one set of calculations to the next and estimate the uncertainty in the final, or absolute, result; however, conducting such a full-scale quantitative uncertainty analysis is often impractical and sometimes impossible, especially for actions to be initiated at an unspecified time in the future. Instead, the risk analysis is designed to ensure, through uniform and judicious selection of scenarios, models, and input parameters, that relative comparisons of risk among the various alternatives are meaningful. In the transportation risk assessment, this design is accomplished by uniformly applying common input parameters and assumptions to each alternative. Therefore, although considerable uncertainty is inherent in the absolute magnitude of the transportation risk for each alternative, much less uncertainty is associated with the relative differences among the alternatives in a given measure of risk.

In the following sections, areas of uncertainty are discussed for the assessment steps enumerated above. Special emphasis is placed on identifying whether the uncertainties affect relative or absolute measures of risk. The degree of reality conservatism of the assumption is addressed. Where practical, the parameters that most significantly affect the risk assessment results are identified.

E.10.1 Uncertainties in Spent Nuclear Fuel Inventory and Characterization

The spent nuclear fuel inventories (i.e., number of shipments) and the physical and radiological characteristics are important input parameters to the transportation risk assessment. The potential amount of transportation for any alternative is determined primarily by the projected spent nuclear fuel inventory and assumptions concerning shipment capacities. The physical and radiological characteristics are important in determining the amount of material released during accidents and the subsequent doses to exposed individuals through multiple environmental exposure pathways.

The development of projected spent nuclear fuel inventory and characterization data used to support the EIS is described in Appendix B. Uncertainties in the spent nuclear fuel inventory and characterization will be reflected to some degree in the transportation risk results. If the spent nuclear fuel inventory (number of elements) is overestimated (or underestimated), the resulting transportation risk estimates also will be overestimated (or underestimated) by roughly the same factor. However, the same spent nuclear fuel inventory estimates are used to analyze the transportation impacts of each of the EIS alternatives. Therefore, for comparative purposes, the observed differences in transportation risks among alternatives are believed to represent unbiased, reasonably accurate estimates from current information in terms of relative risk comparisons.

The spent nuclear fuel type selected for the accident risk calculations was chosen to maximize the potential accident risk results. All accidents were analyzed for fuel that is less than 1 year old. However, much of the fuel has already been out of the foreign research reactors for more than 1 year and may not be brought back for several years. For calculations of MEIs, the cask loaded with the maximum possible amount of radioactive material should have been and was considered. However, the risk values were calculated under the assumption that all casks were loaded to this maximum value. Depending on the implementation of the program, very few, if any, of the casks would be carrying fuel as new as that used in the accident analysis. Selection of another spent nuclear fuel type, or consideration of all spent nuclear fuel types in detail, would result in accident risks less than those reported in the assessment of alternatives in this appendix.

E.10.2 Uncertainties in Casks, Shipment Capacities and Number of Shipments

The amount of transportation required for each alternative is based in part on assumptions concerning the packaging characteristics and shipment capacities for truck and rail modes. Representative shipment capacities have been defined for assessment purposes based on probable future shipment capacities. In reality, the actual shipment capacities may differ from the predicted capacities, so that the projected number of shipments, and consequently the total transportation risk, would change. However, although the predicted transportation risks would increase or decrease accordingly, the relative differences in risks among alternatives would remain about the same. It is in fact likely that DOE would deploy a large capacity truck or rail cask for large intersite shipping campaigns.

For the purposes of analysis, Phase 1 was assumed to last exactly 10 years and Phase 2 was assumed to last exactly 3 years. Realistically, the Phase 2 site may be ready somewhat sooner or later. Additionally, the fractions of the fuel arriving during each phase may not be precisely proportional to the duration of the phase. However, the risk changes are small when compared with the conservatism introduced in the radiological calculations.

The number of shipments to and from various points comes from a complex series of models of how the policy may be implemented. They are not intended to define how the policy would be implemented. Instead, they describe somewhat generally how the policy would be implemented. The risk factors for all conceivable routes between DOE sites and ports of entry are given to show that a slight deviation from the shipment pattern modeled could have a negligible affect on risk. For example, if the policy were being implemented with fuel arriving in the eastern United States going to Savannah River Site and fuel arriving in the western United States going to Idaho National Engineering Laboratory, the risk impact of transporting a few casks from eastern ports to Idaho National Engineering Laboratory (presumably for onsite logistical reasons) would have a small impact on program risk.

E.10.3 Uncertainties in Route Determination

Representative routes have been determined between all origin and destination sites considered in the EIS. The routes have been determined consistent with current guidelines, regulations, and practices, but may not be the actual routes that would be used in the future. In reality, the actual routes could differ from the representative ones in terms of distances and total population along the routes. Moreover, in that the assessment considers spent nuclear fuel could be transported over an extended period of time starting at some time in the future, the highway and rail infrastructures and the demographics along routes could change. These effects have not been accounted for in the transportation assessment, however, it is not anticipated that these changes would significantly affect relative comparisons of risk among the alternatives considered in the EIS.

E.10.4 Uncertainties in the Calculation of Radiation Doses

The models used to calculate radiation doses from transportation activities introduce a further uncertainty in the risk assessment process. It is generally difficult to estimate the accuracy or absolute uncertainty of the risk assessment results. The accuracy of the calculated results is closely related to the limitations of the computational models and to the uncertainties in each of the input parameters that the model requires. The single greatest limitation facing users of RADTRAN, or any computer code of this type, is the scarcity of data for certain input parameters.

Uncertainties associated with the computational models are minimized by using state-of-the-art computer codes that have undergone extensive review. Because there are numerous uncertainties that are recognized but difficult to quantify, assumptions are made at each step of the risk assessment process that are intended to produce conservative results (i.e., overestimate the calculated dose and radiological risk). Because parameters and assumptions are applied equally to all alternatives, this model bias is not expected to affect the meaningfulness of relative comparisons of risk; however, the results may not represent risks in an absolute sense.

In order to understand the most important uncertainties and conservatism in the transportation risk assessment, the results for all cases were examined to identify the largest contributors to the collective population risk. The results of this examination are discussed briefly below.

For truck shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to members of the public at stops, 2) incident-free dose to transportation crew members, 3) incident-free dose to members of the public sharing the route (on-link dose), 4) incident-free dose to members of the public residing along the route (off-link dose), and 5) accident dose risk to members of the public. Approximately 80 percent of the estimated public dose was incurred at stops, 15 percent by the on-link population, and 5 percent by the off-link population. In general, the accident contribution to the total risk was negligible compared with the incident-free risk.

For rail shipments, the largest contributors to the collective population dose were found to be, in decreasing order of importance: 1) incident-free dose to transportation crew members, 2) incident-free dose to members of the public residing along the route (off-link dose), 3) incident-free dose to members of the public at stops, 4) incident-free dose to members of the public sharing the route (on-link dose), and 5) accident dose risk to members of the public. Approximately 70 percent of the estimated public dose was incurred by the off-link population, 25 percent by the population at stops, and 5 percent by the on-link population. As with truck shipments, the accident contribution to the total risk in general was negligible compared with the incident-free risk, even when the spent nuclear fuel type was selected to maximize the accident risk results.

As shown above, incident-free transportation risks are the dominant component of the total transportation risk for both truck and rail modes. The most important parameter in calculating incident-free doses is the shipment external dose rate (incident-free doses are directly proportional to the shipment external dose rate). For this assessment, it was assumed that all shipments would have an external dose rate at the regulatory limit of 10 mrem per hr at 2 m. In practice, the external dose rates would vary from shipment to shipment. Although it is conceivably possible to load a cask with enough fresh foreign research reactor spent nuclear fuel to obtain a dose rate equal to the regulatory limit, experience has shown this to be unlikely. In fact, the observed average dose rate described in Appendix B is approximately ten times lower than the regulatory limit. During the shipments of foreign research reactor to MOTSU and ultimately to Savannah River Site, the State of North Carolina detected less than 1 mrem on contact with the cask and no radiation above background at 2 m (Massey, 1994). Therefore, the incident-free risks are conservative, and would be ten times lower if calculated with the observed average dose.

Finally, the single largest contributor to the collective population doses calculated with RADTRAN was found to be the dose to members of the public at truck stops. Currently, RADTRAN uses a simple point-source approximation for truck-stop exposures and assumes that the total stop time for a shipment is proportional to the shipment distance. The parameters used in the stop model were based on a survey of a very limited number of radioactive material shipments that examined a variety of shipment types in different areas of the country (Wilmot, 1981). It was assumed that stops occur as a function of distance, with a stop rate of 0.011 h per km (0.018 h per mile). It was further assumed that at each stop, an average of 50 people are exposed at a distance of 20 m (66 ft). In RADTRAN, the population dose is directly proportional to the external shipment dose rate and the number of people exposed, and inversely proportional to the square of the distance. The stop rate assumed results in an hour of stop time per 100 km (62 miles) of travel.

Based upon the qualitative discussion with shippers of spent nuclear fuel, the parameter values used in the assessment appear to be conservative. However, data do not exist to qualitatively assess the degree of conservatism in the stop-dose model. As a practical matter, it is conceivable that DOE could take steps to control the location, frequency, and duration of truck stops if necessary. However, based on the regulatory requirements for continuous escort of the material (10 CFR 73) and the requirement for two drivers, it is clear that the trucks would be on the move essentially one-hundred percent of the time until arrival at the destination. Therefore, the calculated impacts are extremely conservative. By using these conservative parameters, the calculations in this EIS are consistent with the RADTRAN default values and the SNF&INEL Final EIS (DOE, 1995).

Shielding of exposed populations is not considered. For all incident-free exposure scenarios, no credit has been taken for shielding of exposed individuals. In reality, shielding would be afforded by trucks and cars sharing the transport routes, natural topography, and the houses and buildings in which people reside. Incident-free exposures to external radiation could be reduced significantly depending on the type of shielding present. For residential houses, shielding factors (i.e., the ratio of shielded to unshielded exposure rates) have been estimated to range from 0.02 to 0.7, with a recommended value of 0.33. If shielding were to be considered for the maximally exposed resident living near a transport route, the calculated doses and risks would be reduced by approximately 70 percent. Similar levels of shielding may be provided to individuals exposed in vehicles. However, consideration of shielding does not significantly affect the overall incident-free risks to the general population.

Post-accident mitigative actions are not considered for dispersal accidents. For severe accidents involving the release and dispersal of radioactive materials in the environment, no post-accident mitigative actions, such as interdiction of crops or evacuation of the accident vicinity, have been considered in this risk assessment. In reality, mitigative actions would take place following an accident in accordance with U.S. Environmental Protection Agency radiation protection guides for nuclear incidents (EPA, 1991). The effects of mitigative actions on population accident doses are highly dependent upon the spent nuclear fuel type involved and the severity, location, and timing of the accident. For this risk assessment, ingestion doses are only calculated for accidents occurring in rural areas (the calculated ingestion dose, however, assumes all food grown on contaminated ground is consumed and is not limited to the rural population).

Examination of the severe accident consequence assessment results has shown that ingestion of contaminated foodstuffs contributes on the order of 50 percent of the total population dose for rural accidents. Interdiction of foodstuffs would act to reduce, but not eliminate, this contribution.

References

DOE (U.S. Department of Energy), 1995, *Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement,* DOE/EIS-0203-F, April.

DOE (U.S. Department of Energy), 1988a, *U.S. Department of Energy, External Dose-Rate Conversion Factors for Calculation of Dose to the Public,* DOE/EH-0070, Washington, DC.

DOE (U.S. Department of Energy), 1988b, *Internal Dose-Rate Conversion Factors for Calculation of Dose to the Public,* DOE/EH-0071, Washington, DC.

DOE (U.S. Department of Energy), 1987, *Analysis of Radiation Doses from Operation of Postulated Commercial Spent Fuel Transportation Systems*, DOE-CH/TPO-001, Pacific Northwest Laboratory, Richland, WA, November.

DOT (U.S. Department of Transportation), 1992, *Guidelines for Selecting Preferred Highway Routes for Highway Route Controlled Quantity Shipments of Radioactive Materials,* DOE/RSPA/HMS/92-02, August.

Doty, S. R., B. L. Wallace, and G. C. Holzworth, 1976, *A Climatological Analysis of Pasquill Stability Categories Based on 'STAR' Summaries,* National Oceanic and Atmospheric Administration, National Climatic Center, Ashville, NC, April.

EPA (U.S. Environmental Protection Agency), 1991, *Manual of Protective Actions for Nuclear Incidents,* EPA/520/1-75-001-A, Washington, DC.

Fischer, L. E., C. K. Chou, M. A. Gerhard, C. Y. Kimura, R. W. Martin, R. W. Mensing, M. E. Mount, and M. C. Wette, 1987, *Shipping Container Response to Severe Highway and Railway Accident Conditions,* NUREG/CR-4829, UCID-20733, prepared by Lawrence Livermore National Laboratory for Division of Reactor System Safety, Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, Washington, DC.

Hutchison, B. C., 1986, *Barge Collisions, Rammings, Groundings; An Engineering Assessment of the Potential for Danger to Radioactive Material Transport Casks,* SAND85-7165, Sandia National Laboratory, Albuquerque, NM, January.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection,* Annals of the ICRP, Publication 60, Vol. 21, No. 1-3, Pergamon Press, New York, November.

Johnson, P. E., D. S. Joy, D. B. Clark, and J. M. Jacobi, 1993a, *HIGHWAY 3.1, An Enhanced Transportation Routing Model: Program Description, Methodology, and Revised User's Manual,* ORNL/TM-12124, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Johnson, P. E., D. S. Joy, D. B. Clark, and J. M. Jacobi, 1993b, *INTERLINE 5.0, An Expanded Railroad Routing Model: Program Description, Methodology, and Revised User's Manual,* ORNL/TM-12090, Oak Ridge National Laboratory, Oak Ridge, TN, March.

Massey, C. D., 1994, Declaration of Charles D. Massey in the U.S. District Court for the District of South Carolina Columbia Division, State of South Carolina vs. Hazel R. O'Leary, Secretary of Energy and the U.S. Department of Energy, Case No. 3-94-2419-0.

Neuhauser, K. S., and F. L. Kanipe, 1993, *RADTRAN 4 Volume II: Technical Manual,* SAND89-2370, Sandia National Laboratories, Albuquerque, NM, August.

NRC (U.S. Nuclear Regulatory Commission), 1995, *Nuclear Regulatory Commission, 10 CFR Part 71, Compatibility with International Atomic Energy Agency (IAEA),* Federal Register, Vol. 60, No. 188, September 28.

NRC (U.S. Nuclear Regulatory Commission), 1993, *Public Informational Circular for Shipments of Irradiated Reactor Fuel*, NUREG-0725, Rev. 9, Washington, DC, March.

NRC (U.S. Nuclear Regulatory Commission), 1977a, *Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes*, NUREG-0170, Volumes 1 and 2, Washington, DC, December.

NRC (U.S. Nuclear Regulatory Commission), 1977b, *Regulatory Guide 1.109 - Calculations of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*, RG 1.109, October.

Rao, R. K., E. L. Wilmot and R. E. Luna, 1982, *Non-Radiological Impacts of Transporting Radioactive Materials*, SAND81-1703, Sandia National Laboratories, Albuquerque, NM, February.

Saricks, C. and T. Kvitek, 1994, *Longitudinal Review of State-Level Accident Statistics for Carriers of Intrastate Freight*, ANL/ESD/TM-68, Argonne National Laboratory, Argonne, IL, March.

Wilmot, E. L., 1981, *Transportation Accident Scenarios for Commercial Spent Fuel*, SAND80-2124, Sandia National Laboratories, Albuquerque, NM, February.

Yuan, Y. C., S. Y. Chen, D. J. LePoire, and R. Rothman, 1993, *RISKIND — A Computer Program for Calculating Radiological Consequences and Health Risks from Transportation of Spent Nuclear Fuel*, ANL/EAIS-06, Rev. 0, Argonne National Laboratory, Argonne, IL, February.

Attachment E1
Representative Routes for Overland Transportation

The purpose of this Attachment is to show the representative routes that are used in the risk analysis of overland transportation of foreign research reactor spent nuclear fuel. Identification of these representative routes is necessary to carry out the risk analysis described in Appendix E. The characteristics of these representative routes are used to estimate the driving time, which is needed to estimate the risk to the crew operating the truck, train, or barge and to estimate population along the route, which is exposed to small amounts of radiation emanating from the cask and could conceivably be exposed to an accident.

Representative routes between all potential ports and Canadian border crossings and all 5 DOE sites, as well as between the DOE sites, were identified. Because of the large number of routes, complexity of routes and the difficulty associated with manually tabulating population data, DOE has developed computer codes to select routes and tabulate populations. The HIGHWAY (Johnson, 1993a) code was used for the road route analysis and the INTERLINE (Johnson, 1993b) code was used for the rail and barge route analysis. The codes are described in Section E.4.

Representative routes are used in the analysis because specific routes cannot be identified in advance. A transportation route is not finalized until it has been reviewed and approved by the NRC (Massey, 1994). Therefore, the routes identified in this section are representative of the routes that may be used to ship spent nuclear fuel, but the actual routes shown may or may not be used if the policy were to be implemented. The codes used to select the representative routes automatically select preferred routes and minimize the transport time, as required by regulation. However, it is required that route selection also considers information such as the time of day and the day of week during which transportation will occur, current conditions such as adverse weather conditions (e.g., flooding, snow), track or road conditions, bridge closures, and seasonal traffic. Additionally, numerous U.S. interstates serve the Canadian border and could be used to transport spent nuclear fuel. HIGHWAY and INTERLINE cannot take these environmental, and other, conditions into account.

Specific requirements for the selection of road routes are found in 49 CFR 397, Subpart D. This regulation requires selection of "preferred routes", which are defined as Interstate System highways for which an alternative route is not designated by a State routing agency and/or State-designated routes selected by a State routing agency. 49 CFR 397 also gives specific directions for the selection of a route between a point of pickup or delivery (i.e., a port facility) and a preferred route. In some cases, the HIGHWAY code does not explicitly cover this short distance. These stretches of road were evaluated, and it was determined that they were very short compared to the overland routes.

This attachment provides maps and listings of the representative routes used in the transportation analysis. Information on population can be found in Tables E-3 and E-4. For each port of entry, a road map followed by a listings of the routes, and a rail map, followed by listings of the routes are provided.

The route listings are from the HIGHWAY and INTERLINE computer code output files. The columns of the HIGHWAY output, from left to right indicate the following: 1) The roadway on which the truck would be traveling, 2) codified information describing the road (# - toll bridge, $ - toll road) and other numbers used to identify the stretch of road, 3) the city nearest the node, 4) the position of the node relative to the city, 5) the intersecting roads that define the node, and 6) the state in which the node is located.

The INTERLINE output, from left to right, shows the following: 1) the railroad that owns a section of the track or, in the case of barge transportation, the letters BRG, 2) the INTERLINE code number for the node, 3) the city nearest the node, 4) the state in which the node is located, and 5) the distance traveled since the last node. Transfers between rail carriers are noted by dashed lines.

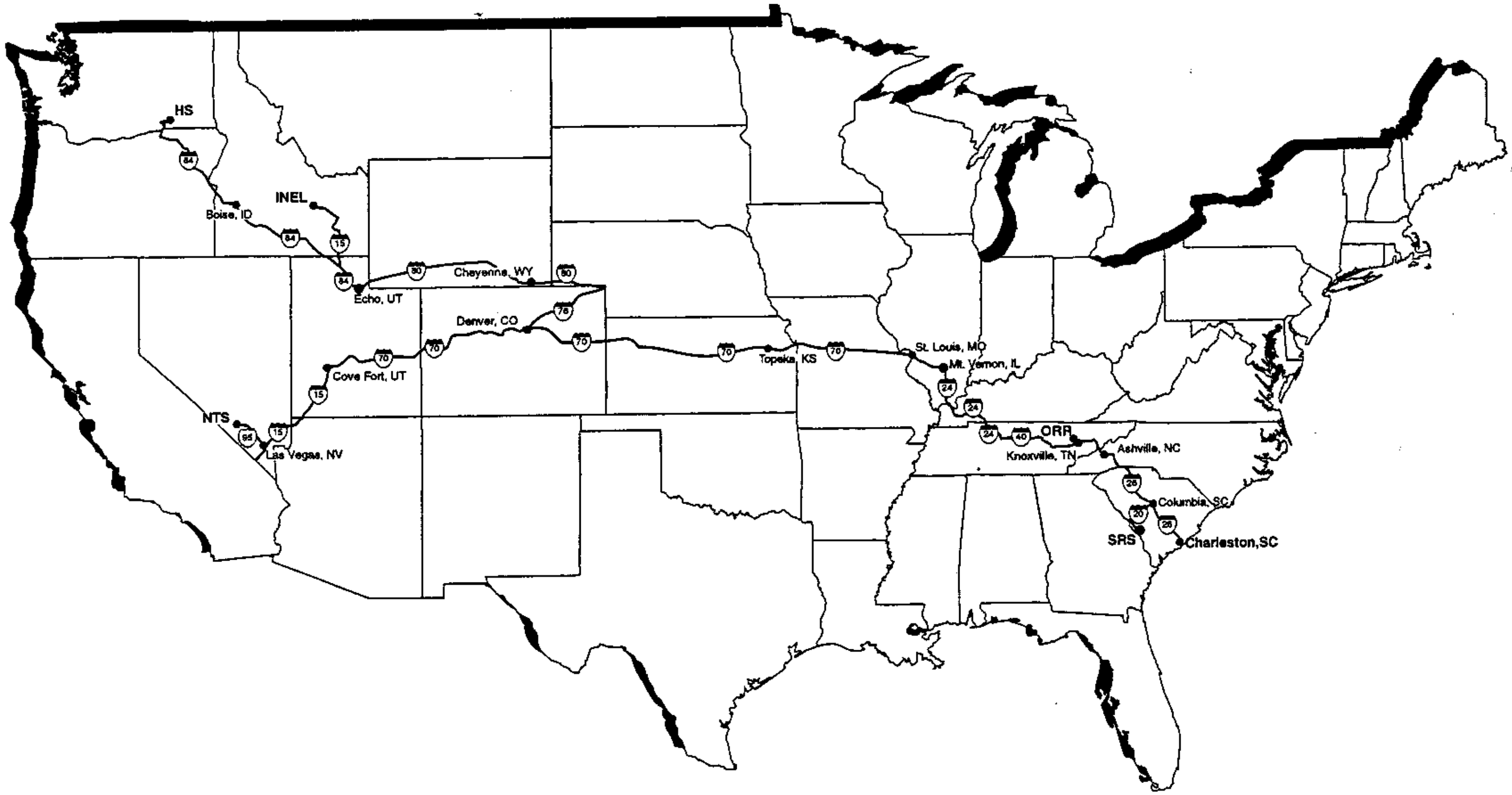


Figure E1-1 Representative Truck Routes from Charleston, SC Area Ports (Charleston-NWS and Charleston-Wando) to Department of Energy Management Sites

From: NWS CHARLESTON, SC
To : SRP BARCD 1

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		BELVEDERE	N	I20	X5	SC
U25		NORTH AUGUSTA	SE	U25	S125	SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		SRP BARCD 1		S125	LC	SC

From: NWS CHARLESTON, SC
To : ID NATL ENG LAB, ID

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U20	LOCL	ID

From: NWS CHARLESTON, SC
To : HANFORD, WA

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NWS CHARLESTON, SC
To : K-25, TN

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: NWS CHARLESTON, SC
To : MERCURY, NV

Routing through:

		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: CHARLESTON (WANDO TERMINAL), SC
To : SRP BARCD 1, SC

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		BELVEDERE	N	I20	X5	SC
U25		NORTH AUGUSTA	SE	U25	S125	SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		SRP BARCD 1		S125	LC	SC

From: CHARLESTON (WANDO TERMINAL), SC
To : ID NATL ENG LAB, ID

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U20	LOCL	ID

From: CHARLESTON (WANDO TERMINAL), SC
To : HANFORD, WA

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: CHARLESTON (WANDO TERMINAL), SC
To : K-25, TN

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: CHARLESTON (WANDO TERMINAL), SC
To : MERCURY, NV

Routing through:

		CHARLESTON	E	I526	X32	SC
I526		N CHARLESTON	NW	I26	I526	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

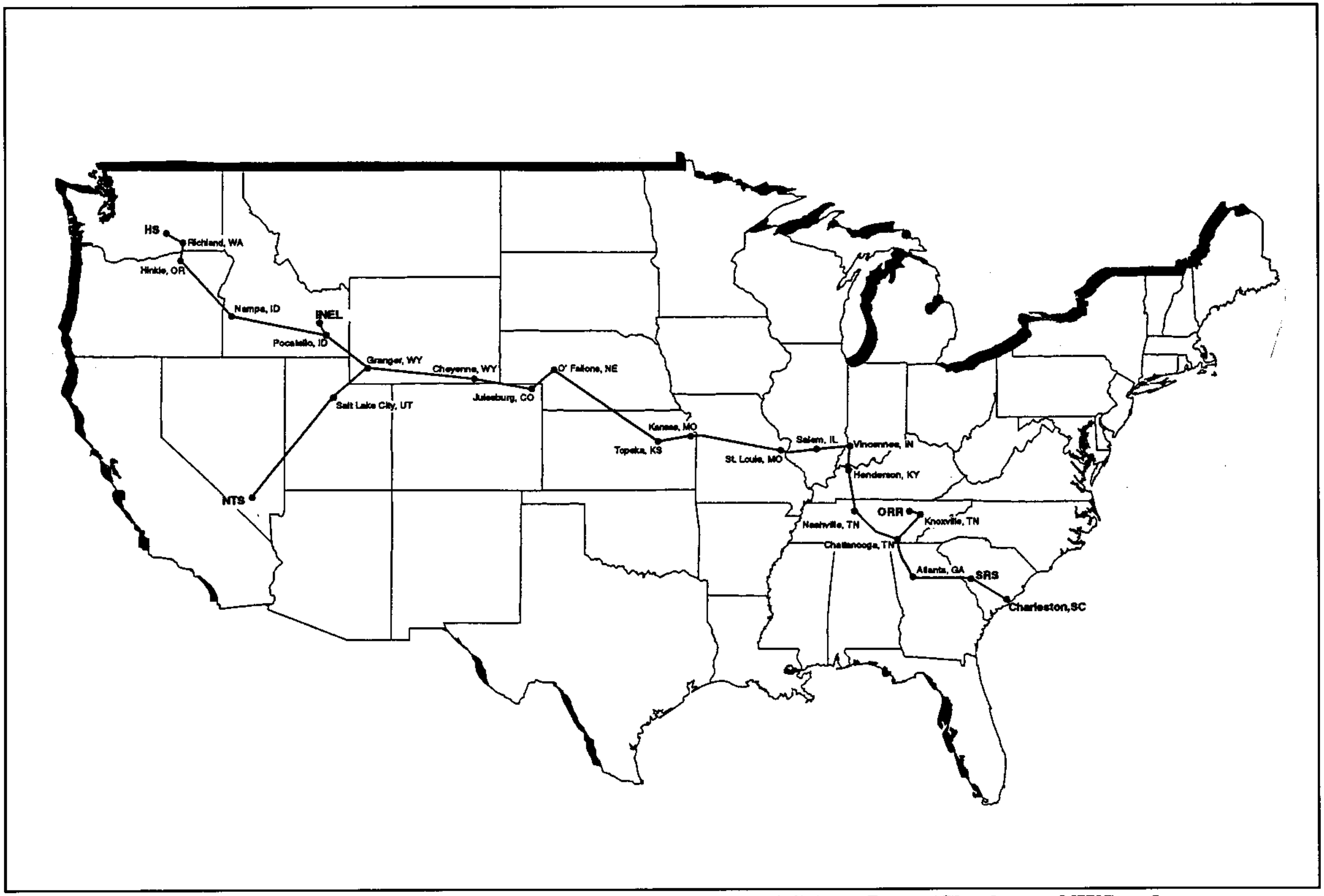


Figure E1-2 Representative Rail Routes from Charleston, SC Area Ports (Charleston-NWS and Charleston-Wando) to Department of Energy Management Sites

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: SRP, SC

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7717-DUNBARTON / WELL	SC	132.
USG	7717-DUNBARTON / WELL	SC	132.
USG	15359-SRP	SC	140.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MADISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	10858-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GIBBON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULESBURG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13369-MC CAMMON	ID	2435.
UP	13370-POCATELLO	ID	2458.
UP	13336-SCOVILLE	ID	2514.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MADISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	10858-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GIBBON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULESBURG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13369-MC CAMMON	ID	2435.
UP	13370-POCATELLO	ID	2458.
UP	13412-NAMPA	ID	2700.
UP	14220-PENDLETON	OR	2968.
UP	14223-HINKLE	OR	2999.
UP	13894-WALLULA	WA	3028.
UP	13964-KENNEWICK	WA	3043.
UP	13941-RICHLAND	WA	3052.
USG	13941-RICHLAND	WA	3052.
USG	16212-HANFORD S 300	WA	3060.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7675-FLORENCE	SC	98.
CSXT	7671-DILLON	SC	127.
CSXT	7470-HAMLET	NC	165.
CSXT	7472-WADESBORO	NC	190.
WSS	7472-WADESBORO	NC	190.
WSS	7462-LEXINGTON	NC	258.
NS	7462-LEXINGTON	NC	258.
NS	7478-SALISBURY	NC	275.
NS	7394-HICKORY	NC	332.
NS	7387-MARION	NC	374.
NS	7343-ASHEVILLE	NC	414.
NS	7318-MORRISTOWN	TN	494.
NS	7286-KNOXVILLE	TN	535.
NS	7288-DOSSETT	TN	560.
NS	15316-K-25	TN	581.

ROUTE FROM: CSXT 7690-CHARLESTON, SC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	7690-CHARLESTON	SC	0.
CSXT	7739-FAIRFAX	SC	94.
CSXT	7732-ROBBINS	SC	123.
CSXT	7961-AUGUSTA	GA	152.
CSXT	7914-ATLANTA	GA	327.
CSXT	7907-MARIETTA	GA	337.
CSXT	7889-CARTERSVILLE	GA	369.
CSXT	7888-DALTON	GA	420.
CSXT	7235-CHATTANOOGA	TN	458.
CSXT	7187-TULLAHOMA	TN	539.
CSXT	7202-NASHVILLE	TN	618.
CSXT	7201-MADISON	TN	628.
CSXT	7061-HOPKINSVILLE	KY	688.
CSXT	3839-HENDERSON	KY	775.
CSXT	3838-EVANSVILLE	IN	788.
CSXT	3812-VINCENNES	IN	838.
CSXT	4952-SALEM	IL	917.
CSXT	10859-EAST ST LOUIS	IL	982.
<TR>	10859-EAST ST LOUIS	IL	982.
<TR>	10858-ST LOUIS	MO	988.
UP	10858-ST LOUIS	MO	988.
UP	10656-JEFFERSON CITY	MO	1110.
UP	10616-KANSAS CITY	MO	1286.
UP	10617-KANSAS CITY	KS	1289.
UP	11823-LAWRENCE	KS	1328.
UP	11697-TOPEKA	KS	1358.
UP	11696-MENOKEN	KS	1363.
UP	11681-MARYSVILLE	KS	1438.
UP	11405-HASTINGS	NE	1548.
UP	11410-GIBBON	NE	1574.
UP	11352-NORTH PLATTE	NE	1652.
UP	11358-O FALLONS	NE	1701.
UP	13703-JULESBURG	CO	1769.
UP	13465-CHEYENNE	WY	1915.
UP	13462-LARAMIE	WY	1967.
UP	13494-GRANGER	WY	2243.
UP	13568-OGDEN	UT	2382.
UP	13595-SALT LAKE CITY	UT	2417.
UP	13630-LYNNDYL	UT	2529.
UP	14766-VALLEY	NV	2846.
USG	14766-VALLEY	NV	2846.
USG	16333-YUCCA MOUNTAIN	NV	2945.

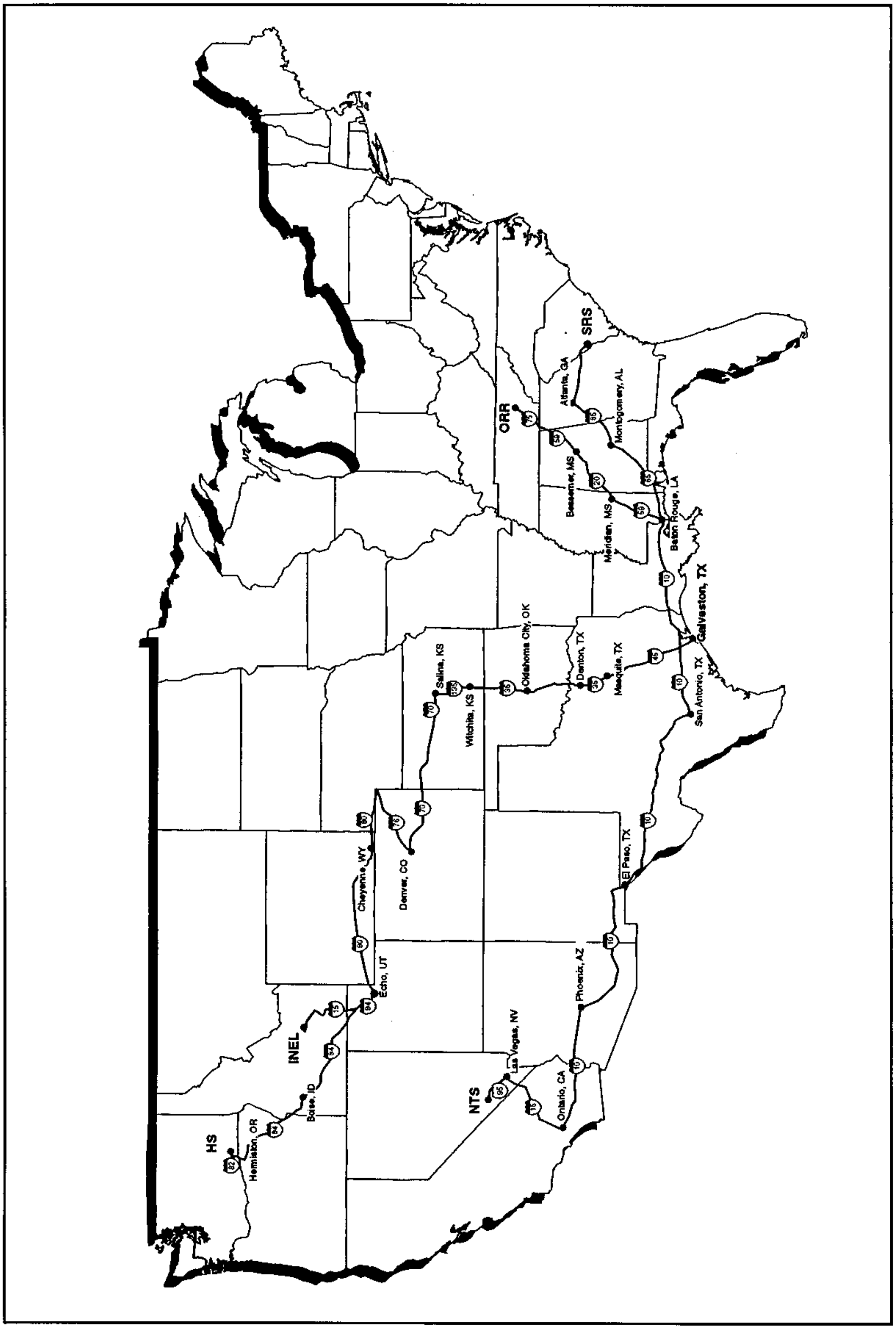


Figure E1-3 Representative Truck Routes from Galveston, TX to Department of Energy Management Sites

From: GALVESTON, TX
To : ID NATL ENG LAB

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	N	I45	I610	TX
I45		DALLAS	SE	I45	I635	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I635		DALLAS	NW	I35E	I635	TX
I35E		DENTON	S	I35E	I35W	TX
I35		OKLAHOMA CITY	S	I240	I35	OK
I240		OKLAHOMA CITY	SW	I240	I44	OK
I44		OKLAHOMA CITY	NE	I35	I44	OK
I35	I44	EDMOND	SE	I35	I44	OK
I35		SOUTH HAVEN	E	I35	U166	KS
I35 $	TKST$	WICHITA	S	I235	I35	KS
I235		WICHITA	N	I135	I235	KS
I135		SALINA	NW	I135	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: GALVESTON, TX
To : HANFORD, WA

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	N	I45	I610	TX
I45		DALLAS	SE	I45	I635	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I635		DALLAS	NW	I35E	I635	TX
I35E		DENTON	S	I35E	I35W	TX
I35		OKLAHOMA CITY	S	I240	I35	OK
I240		OKLAHOMA CITY	SW	I240	I44	OK
I44		OKLAHOMA CITY	NE	I35	I44	OK
I35	I44	EDMOND	SE	I35	I44	OK
I35		SOUTH HAVEN	E	I35	U166	KS
I35 $	TKST$	WICHITA	S	I235	I35	KS
I235		WICHITA	N	I135	I235	KS
I135		SALINA	NW	I135	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: GALVESTON, TX
To : K-25, TN

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	E	I10	I610	TX
I10		BATON ROUGE	SE	I10	I12	LA
I12		SLIDELL	NE	I10	I12	LA
I59		MERIDIAN	W	I20	I59	MS
I20	I59	BESSEMER	SW	I20	I459	AL
yI459		CENTER POINT	SE	I459	I59	AL
I59		WILDWOOD	NW	I24	I59	GA
I24		EAST RIDGE	NE	I24	I75	TN
I75		OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: GALVESTON, TX
To : MERCURY, NV

Routing through:

		GALVESTON				TX
I45		HOUSTON	SE	I45	I610	TX
I610		HOUSTON	W	I10	I610	TX
I10		SAN ANTONIO	E	I10	I410	TX
I410		KIRBY	N	I35	I410	TX
I35	I410	SAN ANTONIO	NE	I35	I410	TX
I410		SAN ANTONIO	NW	I10	I410	TX
I10		PHOENIX	SE	I10	I17	AZ
I17		PHOENIX	W	I10	I17	AZ
I10		ONTARIO	E	I10	I15	CA
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

Figure E1-4 Representative Rail Routes from Galveston, TX to Department of Energy Management Sites

Route from: ATSF 13047-GALVESTON, TX
To: USG 15359-SRP, SC

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12399-HOUSTON	TX	68.
<TR>	12399-HOUSTON	TX	68.
UP	12399-HOUSTON	TX	68.
UP	12341-BEAUMONT	TX	156.
UP	9122-DE QUINCY	LA	203.
UP	9101-KINDER	LA	239.
UP	9112-LIVONIA	LA	317.
UP	8985-NEW ORLEANS	LA	446.
CSXT	8985-NEW ORLEANS	LA	446.
CSXT	8966-GULFPORT	MS	520.
CSXT	8926-BILOXI	MS	536.
CSXT	8967-PASCAGOULA	MS	553.
CSXT	8597-MOBILE	AL	592.
CSXT	8566-FLOMATON	AL	644.
CSXT	8657-MONTGOMERY	AL	769.
CSXT	8683-OPELIKA	AL	835.
CSXT	8142-LA GRANGE	GA	878.
CSXT	7914-ATLANTA	GA	954.
CSXT	7961-AUGUSTA	GA	1129.
CSXT	7732-ROBBINS	SC	1158.
CSXT	7717-DUNBARTON / WELL	SC	1167.
USG	7717-DUNBARTON / WELL	SC	1167.
USG	15359-SRP	SC	1175.

Route from: ATSF 13047-GALVESTON, TX
To: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12793-CANYON	TX	686.
ATSF	12792-AMARILLO	TX	703.
ATSF	13753-LA JUNTA	CO	963.
ATSF	13764-PUEBLO	CO	1018.
ATSF	13760-COLORADO SPRINGS	CO	1061.
ATSF	13727-DENVER	CO	1138.
UP	13727-DENVER	CO	1138.
UP	13712-LA SALLE	CO	1181.
UP	13709-GREELEY	CO	1188.
UP	13465-CHEYENNE	WY	1247.
UP	13462-LARAMIE	WY	1299.
UP	13494-GRANGER	WY	1575.
UP	13369-MC CAMMON	ID	1767.
UP	13370-POCATELLO	ID	1790.
UP	13336-SCOVILLE	ID	1846.

Route from: ATSF 13047-GALVESTON, TX
To: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12793-CANYON	TX	686.
ATSF	12792-AMARILLO	TX	703.
ATSF	13753-LA JUNTA	CO	963.
ATSF	13764-PUEBLO	CO	1018.
ATSF	13760-COLORADO SPRINGS	CO	1061.
ATSF	13727-DENVER	CO	1138.
UP	13727-DENVER	CO	1138.
UP	13712-LA SALLE	CO	1181.
UP	13709-GREELEY	CO	1188.
UP	13465-CHEYENNE	WY	1247.
UP	13462-LARAMIE	WY	1299.
UP	13494-GRANGER	WY	1575.
UP	13369-MC CAMMON	ID	1767.
UP	13370-POCATELLO	ID	1790.
UP	13412-NAMPA	ID	2032.
UP	14220-PENDLETON	OR	2301.
UP	14223-HINKLE	OR	2332.
UP	13894-WALLULA	WA	2361.
UP	13964-KENNEWICK	WA	2376.
UP	13941-RICHLAND	WA	2384.
USG	13941-RICHLAND	WA	2384.
USG	16212-HANFORD S 300	WA	2392.

Route From: ATSF 13047-GALVESTON, TX
To: NS 15316-K-25, TN

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12399-HOUSTON	TX	68.
<TR>	12399-HOUSTON	TX	68.
UP	12399-HOUSTON	TX	68.
UP	12341-BEAUMONT	TX	156.
UP	9122-DE QUINCY	LA	203.
UP	9101-KINDER	LA	239.
UP	9112-LIVONIA	LA	317.
UP	8985-NEW ORLEANS	LA	446.
NS	8985-NEW ORLEANS	LA	446.
NS	8986-SLIDELL	LA	476.
NS	8963-HATTIESBURG	MS	558.
NS	8887-MERIDIAN	MS	641.
NS	8640-BOLIGEE	AL	693.
NS	8754-TUSCALOOSA	AL	737.
NS	8739-BIRMINGHAM	AL	807.
NS	8797-GADSDEN	AL	869.
NS	7235-CHATTANOOGA	TN	957.
NS	7260-HARRIMAN	TN	1038.
NS	15316-K-25	TN	1053.

Route from: ATSF 13047-GALVESTON, TX
To: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
ATSF	13047-GALVESTON	TX	0.
ATSF	13054-TEXAS CITY JCT	TX	15.
ATSF	12392-ALVIN	TX	40.
ATSF	12466-SOMERVILLE	TX	153.
ATSF	12480-TEMPLE	TX	230.
ATSF	12414-KILLEEN	TX	260.
ATSF	12725-BROWNWOOD	TX	353.
ATSF	12830-SWEETWATER	TX	468.
ATSF	12812-LUBBOCK	TX	582.
ATSF	12806-FARWELL	TX	672.
ATSF	13025-CLOVIS	NM	683.
ATSF	12995-BELEN	NM	924.
ATSF	12996-DALIES	NM	933.
ATSF	16077-GRANTS	NM	993.
ATSF	12999-GALLUP	NM	1070.
ATSF	12949-HOLBROOK	AZ	1181.
ATSF	12959-FLAGSTAFF	AZ	1276.
ATSF	12964-WILLIAMS	AZ	1305.
ATSF	12963-KINGMAN	AZ	1458.
ATSF	14663-DAGGETT	CA	1677.
UP	14663-DAGGETT	CA	1677.
UP	14762-LAS VEGAS	NV	1841.
UP	14766-VALLEY	NV	1856.
USG	14766-VALLEY	NV	1856.
USG	16333-YUCCA MOUNTAIN	NV	1955.

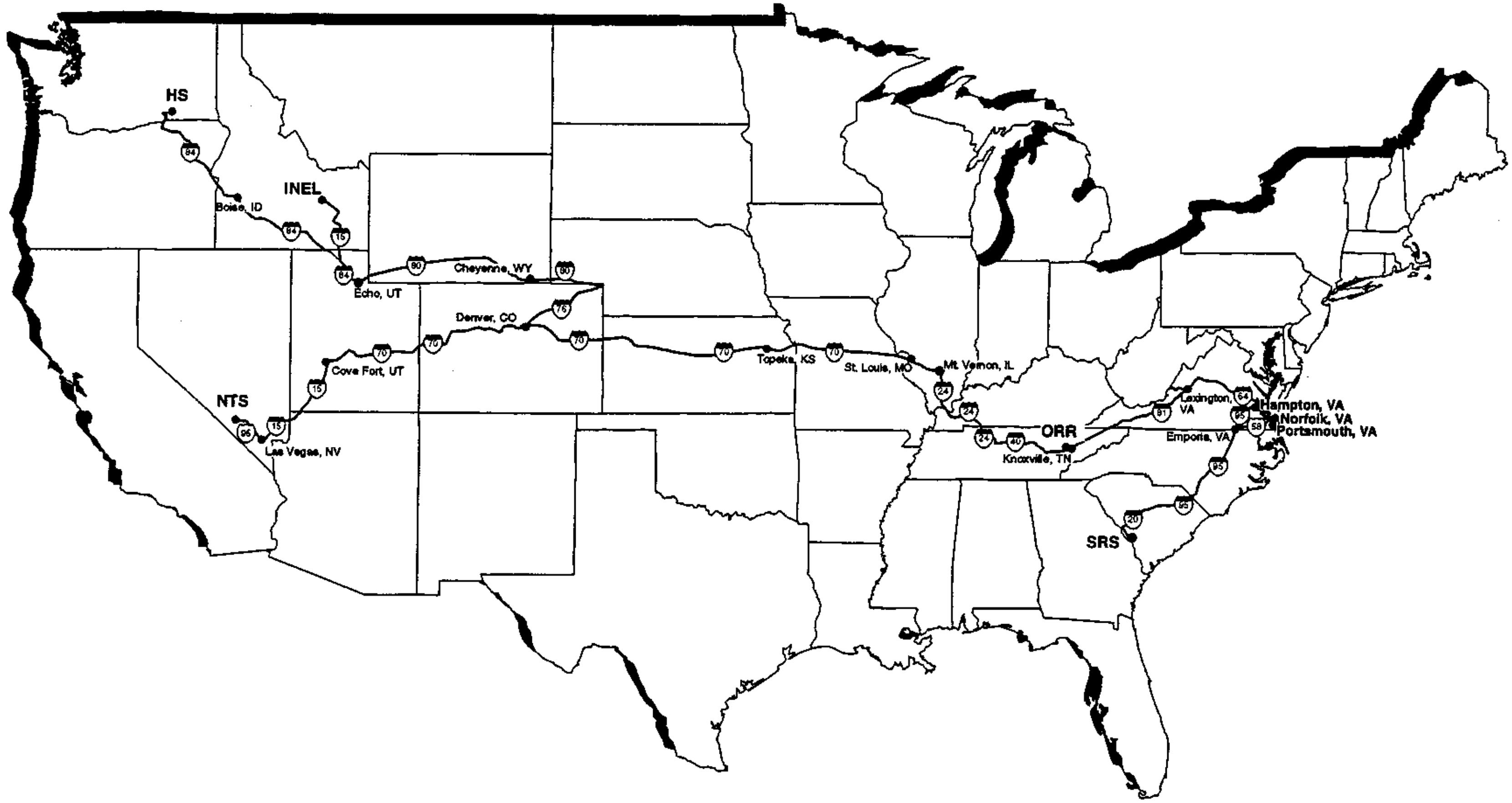


Figure E1-5 Representative Truck Routes from Hampton Roads Area Ports (Newport News, Norfolk, and Portsmouth, VA) to Department of Energy Management Sites

From: NEWPORT NEWS S I664 U60, VA
To : SRP, SC

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		HAMPTON	SE	I64	BRDG	VA
I64 #		WILLOGHBY BCH				VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
I230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: NEWPORT NEWS S I664 U60 VA
To : ID NATL ENG LAB ID

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: NEWPORT NEWS S I664 U60 VA
To : HANFORD, WA

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NEWPORT NEWS S I664 U60 VA
To : K-25, TN

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: NEWPORT NEWS S I664 U60 VA
To : MERCURY, NV

Routing through:

		NEWPORT NEWS	S	I664	U60	VA
I664		HAMPTON		I64	I664	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: NORFOLK E U13 U58 VA
To : SRP, SC

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: NORFOLK, E U13 U58 VA
To : ID NATL ENG LAB, ID

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I270		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: NORFOLK N I564 I64 VA
To : HANFORD, WA

Routing through:

		NORFOLK	N	I564	I64	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: NORFOLK E U13 U58 VA
To : K-25, TN

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I85 $		MATOACA	S	I85	U1	VA
I85		SUTHERLAND	E	I85	U460	VA
U460		BURKEVILLE	E	U360	U460	VA
U360	U460	BURKEVILLE	W	U360	U460	VA
U460		FARMVILLE	S	U15	U460	VA
U15	U460	FARMVILLE	W	U15	U460	VA
U460		BEDFORD				VA
U221	U460	BONSACK		U221	S220	VA
S220A		CLOVERDALE	N	I81	U220	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: NORFOLK E U13 U58 VA
To : MERCURY, NV

Routing through:

		NORFOLK	E	U13	U58	VA
U13		NORFOLK	NE	I64	U13	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U460		PETERSBURG				VA
I95 $	TRPT$	RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I95 $	TRPT$	RICHMOND	N	I95	U301	VA
I95		GLEN ALLEN	E	I295	I95	VA
I295		SHORT PUMP	NE	I295	I64	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: PORTSMOUTH, VA
To : SRP, SC

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		CHESAPEAKE	W	I264	I64	VA
U13	U460	SUFFOLK	E	U13	U460	VA
U460	U58	SUFFOLK	N	U460	U58	VA
U58		EMPORIA	N	I95	U58	VA
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: PORTSMOUTH, VA
To : ID NATL ENG LAB, ID

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: PORTSMOUTH, VA
To : HANFORD, WA

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: PORTSMOUTH, VA
To : K-25, TN

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: PORTSMOUTH, VA
To : MERCURY, NV

Routing through:

		PORTSMOUTH				VA
U460		NORFOLK				VA
I264		NORFOLK	E	I264	I64	VA
I64		WILLOGHBY BCH				VA
I64 #		HAMPTON	SE	I64	BRDG	VA
I64		RICHMOND	N	I64	I95	VA
I64 $	I95 $	RICHMOND	NW	I64	I95	VA
I64		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I64		BECKLEY	S	I64	I77	WV
I64 $	I77 $	CHARLESTON	SE	I64	U60	WV
I64	I77	CHARLESTON		I64	I77	WV
I64		LEXINGTON	E	I64	I75	KY
I64	I75	LEXINGTON	N	I64	I75	KY
I64		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

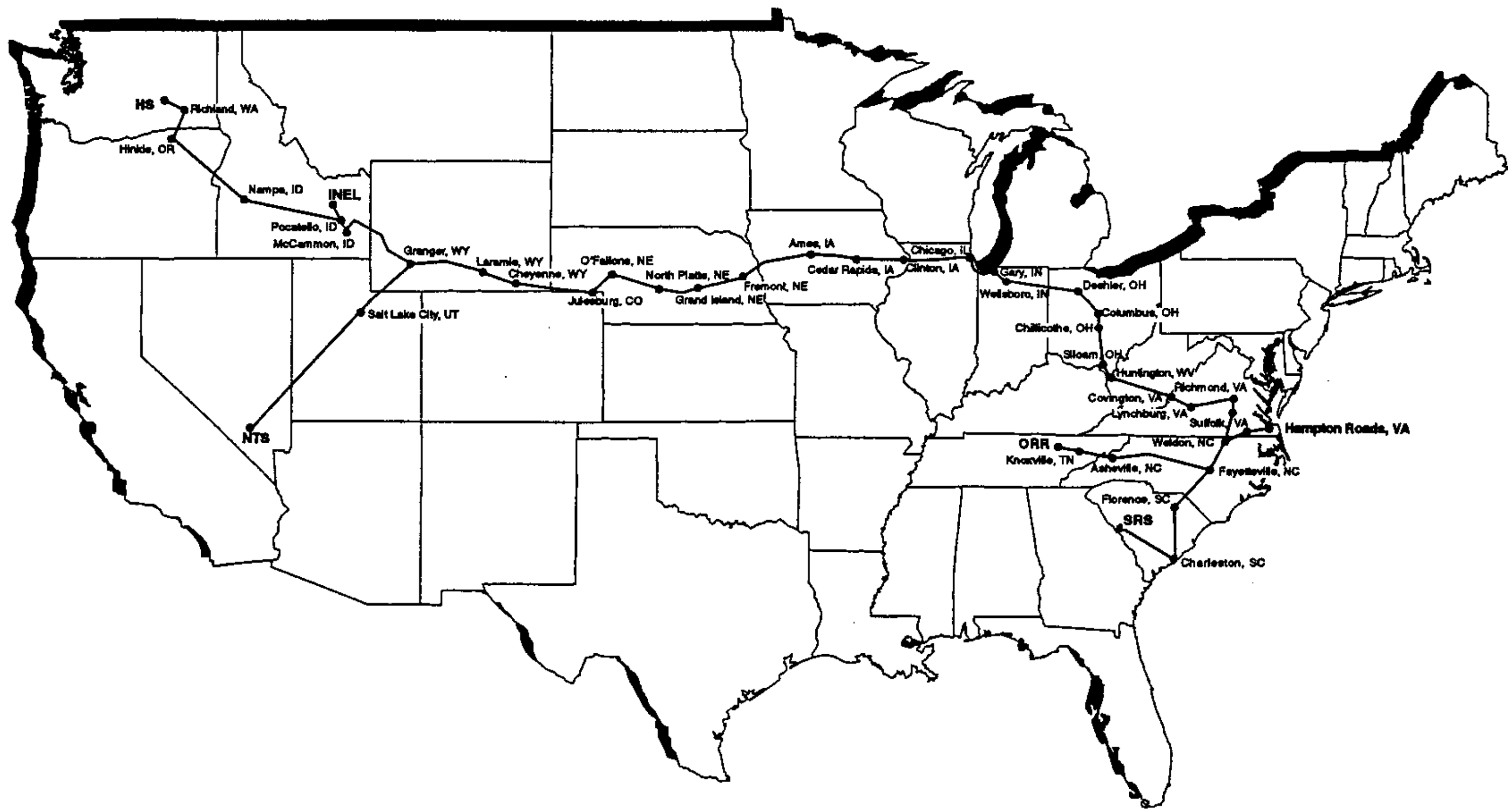


Figure E1-6 Representative Rail Routes from Hampton Roads Area Ports (Newport News, Norfolk, and Portsmouth, VA) to Department of Energy Management Sites

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6087-COLONIAL HEIGHTS	VA	97.
CSXT	6064-PETERSBURG	VA	103.
CSXT	7563-WELDON	NC	163.
CSXT	7565-ROCKY MOUNT	NC	200.
CSXT	7566-WILSON	NC	214.
CSXT	7606-FAYETTEVILLE	NC	288.
CSXT	7620-PEMBROKE	NC	317.
CSXT	7671-DILLON	SC	337.
CSXT	7675-FLORENCE	SC	366.
CSXT	7690-CHARLESTON	SC	464.
CSXT	7739-FAIRFAX	SC	558.
CSXT	7732-ROBBINS	SC	587.
CSXT	7717-DUNBARTON / WELL	SC	596.
USG	7717-DUNBARTON / WELL	SC	596.
USG	15359-SRP	SC	604.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	6806-BARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-BURNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.
CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	11340-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	11358-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13369-MC CAMMON	ID	2463.
UP	13370-POCATELLO	ID	2486.
UP	13336-SCOVILLE	ID	2542.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 16212-HANFORD S 300 WA

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	6806-BARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-BURNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.
CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	11340-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	11358-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13369-MC CAMMON	ID	2463.
UP	13370-POCATELLO	ID	2486.
UP	13412-NAMPA	ID	2728.
UP	14220-PENDLETON	OR	2997.
UP	14223-HINKLE	OR	3028.
UP	13894-WALLULA	WA	3057.
UP	13964-KENNEWICK	WA	3072.
UP	13941-RICHLAND	WA	3080.
USG	13941-RICHLAND	WA	3080.
USG	16212-HANFORD S 300	WA	3088.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6087-COLONIAL HEIGHTS	VA	97.
CSXT	6064-PETERSBURG	VA	103.
CSXT	7563-WELDON	NC	163.
CSXT	7565-ROCKY MOUNT	NC	200.
CSXT	7566-WILSON	NC	214.
CSXT	7606-FAYETTEVILLE	NC	288.
CSXT	7620-PEMBROKE	NC	317.
CSXT	7470-HAMLET	NC	348.
CSXT	7472-WADESBORO	NC	373.
WSS	7472-WADESBORO	NC	373.
WSS	7462-LEXINGTON	NC	441.
NS	7462-LEXINGTON	NC	441.
NS	7478-SALISBURY	NC	458.
NS	7394-HICKORY	NC	515.
NS	7387-MARION	NC	557.
NS	7343-ASHEVILLE	NC	597.
NS	7318-MORRISTOWN	TN	677.
NS	7286-KNOXVILLE	TN	718.
NS	7288-DOSSETT	TN	743.
NS	15316-K-25	TN	764.

ROUTE FROM: CSXT 6024-NEWPORT NEWS, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6024-NEWPORT NEWS	VA	0.
CSXT	6082-RICHMOND	VA	76.
CSXT	6220-LYNCHBURG	VA	209.
CSXT	6198-GLASGOW	VA	235.
CSXT	6200-CLIFTON FORGE	VA	290.
CSXT	6189-COVINGTON	VA	317.
CSXT	6517-PRINCE	WV	384.
CSXT	6794-WEST CHARLESTOWN	WV	460.
CSXT	6795-ST ALBANS	WV	468.
CSXT	6806-BARBOURSVILLE	WV	499.
CSXT	6811-HUNTINGTON	WV	510.
CSXT	6808-KENOVA	WV	516.
CSXT	6807-CATLETTSBURG	KY	518.
CSXT	6809-ASHLAND	KY	524.
CSXT	6846-SILOAM	KY	547.
CSXT	3162-CHILLICOTHE	OH	609.
CSXT	3095-COLUMBUS (BROAD	OH	651.
CSXT	3402-MARION	OH	696.
CSXT	3002-FOSTORIA	OH	739.
CSXT	3484-DESHLER	OH	768.
CSXT	3993-WELLSBORO	IN	916.
CSXT	4070-GARY	IN	950.
CSXT	4073-CLARKE	IN	954.
CSXT	4074-INDIANA HARBOR	IN	957.
CSXT	4232-SOUTH CHICAGO	IL	965.
CSXT	4231-BURNSIDE	IL	968.
CSXT	4217-CHICAGO	IL	980.
CNW	4217-CHICAGO	IL	980.
CNW	4234-PROVISO	IL	994.
CNW	4311-DE KALB	IL	1036.
CNW	4324-NELSON	IL	1081.
CNW	10304-CLINTON	IA	1114.
CNW	10289-CEDAR RAPIDS	IA	1195.
CNW	10265-MARSHALLTOWN	IA	1262.
CNW	10246-NEVADA	IA	1289.
CNW	10271-AMES	IA	1300.

ROUTE FROM : CSXT 6024-NEWPORT NEWS, VA
TO : USG 16333-YUCCA MOUNTAIN, NV -
Continued from Column 1

RR	NODE	STATE	DIST
CNW	10176-MISSOURI VALLEY	IA	1433.
CNW	10198-CALIFORNIA JCT	IA	1439.
CNW	11340-FREMONT	NE	1467.
UP	11340-FREMONT	NE	1467.
UP	11406-GRAND ISLAND	NE	1576.
UP	11410-GIBBON	NE	1602.
UP	11352-NORTH PLATTE	NE	1680.
UP	11358-O FALLONS	NE	1729.
UP	13703-JULESBURG	CO	1797.
UP	13465-CHEYENNE	WY	1943.
UP	13462-LARAMIE	WY	1995.
UP	13494-GRANGER	WY	2271.
UP	13568-OGDEN	UT	2410.
UP	13595-SALT LAKE CITY	UT	2445.
UP	13630-LYNNDYL	UT	2558.
UP	14766-VALLEY	NV	2875.
USG	14766-VALLEY	NV	2875.
USG	16333-YUCCA MOUNTAIN	NV	2974.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13369-MC CAMMON	ID	2562.
UP	13370-POCATELLO	ID	2585.
UP	13336-SCOVILLE	ID	2641.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 16212-HANFORD S 300 WA

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13369-MC CAMMON	ID	2562.
UP	13370-POCATELLO	ID	2585.
UP	13412-NAMPA	ID	2827.
UP	14220-PENDLETON	OR	3096.
UP	14223-HINKLE	OR	3127.
UP	13894-WALLULA	WA	3156.
UP	13964-KENNEWICK	WA	3171.
UP	13941-RICHLAND	WA	3179.
USG	13941-RICHLAND	WA	3179.
USG	16212-HANFORD S 300	WA	3187.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	7565-ROCKY MOUNT	NC	125.
CSXT	7566-WILSON	NC	139.
CSXT	7606-FAYETTEVILLE	NC	213.
CSXT	7620-PEMBROKE	NC	242.
CSXT	7470-HAMLET	NC	273.
CSXT	7472-WADESBORO	NC	298.
WSS	7472-WADESBORO	NC	298.
WSS	7462-LEXINGTON	NC	366.
NS	7462-LEXINGTON	NC	366.
NS	7478-SALISBURY	NC	383.
NS	7394-HICKORY	NC	440.
NS	7387-MARION	NC	482.
NS	7343-ASHEVILLE	NC	522.
NS	7318-MORRISTOWN	TN	602.
NS	7286-KNOXVILLE	TN	643.
NS	7288-DOSSETT	TN	668.
NS	15316-K-25	TN	689.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	7565-ROCKY MOUNT	NC	125.
CSXT	7566-WILSON	NC	139.
CSXT	7606-FAYETTEVILLE	NC	213.
CSXT	7620-PEMBROKE	NC	242.
CSXT	7671-DILLON	SC	262.
CSXT	7675-FLORENCE	SC	291.
CSXT	7690-CHARLESTON	SC	389.
CSXT	7739-FAIRFAX	SC	483.
CSXT	7732-ROBBINS	SC	512.
CSXT	7717-DUNBARTON / WELL	SC	521.
USG	7717-DUNBARTON / WELL	SC	521.
USG	15359-SRP	SC	529.

ROUTE FROM: CSXT 6003-NORFOLK, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6003-NORFOLK	VA	0.
CSXT	6059-SUFFOLK	VA	30.
CSXT	7563-WELDON	NC	88.
CSXT	6064-PETERSBURG	VA	148.
CSXT	6087-COLONIAL HEIGHTS	VA	154.
CSXT	6082-RICHMOND	VA	175.
CSXT	6220-LYNCHBURG	VA	308.
CSXT	6198-GLASGOW	VA	334.
CSXT	6200-CLIFTON FORGE	VA	389.
CSXT	6189-COVINGTON	VA	416.
CSXT	6517-PRINCE	WV	483.
CSXT	6794-WEST CHARLESTOWN	WV	559.
CSXT	6795-ST ALBANS	WV	567.
CSXT	6806-BARBOURSVILLE	WV	598.
CSXT	6811-HUNTINGTON	WV	609.
CSXT	6808-KENOVA	WV	615.
CSXT	6807-CATLETTSBURG	KY	617.
CSXT	6809-ASHLAND	KY	623.
CSXT	6846-SILOAM	KY	646.
CSXT	3162-CHILLICOTHE	OH	709.
CSXT	3095-COLUMBUS (BROAD	OH	751.
CSXT	3402-MARION	OH	796.
CSXT	3002-FOSTORIA	OH	839.
CSXT	3484-DESHLER	OH	867.
CSXT	3993-WELLSBORO	IN	1015.
CSXT	4070-GARY	IN	1049.
CSXT	4073-CLARKE	IN	1053.
CSXT	4074-INDIANA HARBOR	IN	1056.
CSXT	4232-SOUTH CHICAGO	IL	1064.
CSXT	4231-BURNSIDE	IL	1067.
CSXT	4217-CHICAGO	IL	1079.
CNW	4217-CHICAGO	IL	1079.
CNW	4234-PROVISO	IL	1094.
CNW	4311-DE KALB	IL	1136.
CNW	4324-NELSON	IL	1181.
CNW	10304-CLINTON	IA	1213.
CNW	10289-CEDAR RAPIDS	IA	1294.
CNW	10265-MARSHALLTOWN	IA	1361.
CNW	10246-NEVADA	IA	1388.
CNW	10271-AMES	IA	1399.
CNW	10176-MISSOURI VALLEY	IA	1532.
CNW	10198-CALIFORNIA JCT	IA	1538.
CNW	11340-FREMONT	NE	1566.
UP	11340-FREMONT	NE	1566.
UP	11406-GRAND ISLAND	NE	1675.
UP	11410-GIBBON	NE	1701.
UP	11352-NORTH PLATTE	NE	1779.
UP	11358-O FALLONS	NE	1828.
UP	13703-JULESBURG	CO	1896.
UP	13465-CHEYENNE	WY	2042.
UP	13462-LARAMIE	WY	2094.
UP	13494-GRANGER	WY	2370.
UP	13568-OGDEN	UT	2509.
UP	13595-SALT LAKE CITY	UT	2545.
UP	13630-LYNNDYL	UT	2657.
UP	14766-VALLEY	NV	2974.
USG	14766-VALLEY	NV	2974.
USG	16333-YUCCA MOUNTAIN	NV	3073.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	7565-ROCKY MOUNT	NC	95.
CSXT	7566-WILSON	NC	109.
CSXT	7606-FAYETTEVILLE	NC	183.
CSXT	7620-PEMBROKE	NC	212.
CSXT	7671-DILLON	SC	232.
CSXT	7675-FLORENCE	SC	261.
CSXT	7690-CHARLESTON	SC	359.
CSXT	7739-FAIRFAX	SC	453.
CSXT	7732-ROBBINS	SC	482.
CSXT	7717-DUNBARTON / WELL	SC	491.
USG	7717-DUNBARTON / WELL	SC	491.
USG	15359-SRP	SC	499.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	11410-GIBBON	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLONS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13369-MC CAMMON	ID	2532.
UP	13370-POCATELLO	ID	2555.
UP	13336-SCOVILLE	ID	2611.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	11410-GIBBON	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLONS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13369-MC CAMMON	ID	2532.
UP	13370-POCATELLO	ID	2555.
UP	13412-NAMPA	ID	2797.
UP	14220-PENDLETON	OR	3066.
UP	14223-HINKLE	OR	3097.
UP	13894-WALLULA	WA	3126.
UP	13964-KENNEWICK	WA	3141.
UP	13941-RICHLAND	WA	3149.
USG	13941-RICHLAND	WA	3149.
USG	16212-HANFORD S 300	WA	3157.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	7565-ROCKY MOUNT	NC	95.
CSXT	7566-WILSON	NC	109.
CSXT	7606-FAYETTEVILLE	NC	183.
CSXT	7620-PEMBROKE	NC	212.
CSXT	7470-HAMLET	NC	243.
CSXT	7472-WADESBORO	NC	268.
WSS	7472-WADESBORO	NC	268.
WSS	7462-LEXINGTON	NC	336.
NS	7462-LEXINGTON	NC	336.
NS	7478-SALISBURY	NC	353.
NS	7394-HICKORY	NC	410.
NS	7387-MARION	NC	452.
NS	7343-ASHEVILLE	NC	492.
NS	7318-MORRISTOWN	TN	572.
NS	7286-KNOXVILLE	TN	613.
NS	7288-DOSSETT	TN	638.
NS	15316-K-25	TN	659.

ROUTE FROM: CSXT 6059-SUFFOLK, VA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	6059-SUFFOLK	VA	0.
CSXT	7563-WELDON	NC	58.
CSXT	6064-PETERSBURG	VA	118.
CSXT	6087-COLONIAL HEIGHTS	VA	124.
CSXT	6082-RICHMOND	VA	145.
CSXT	6220-LYNCHBURG	VA	278.
CSXT	6198-GLASGOW	VA	304.
CSXT	6200-CLIFTON FORGE	VA	359.
CSXT	6189-COVINGTON	VA	386.
CSXT	6517-PRINCE	WV	453.
CSXT	6794-WEST CHARLESTOWN	WV	529.
CSXT	6795-ST ALBANS	WV	537.
CSXT	6806-BARBOURSVILLE	WV	568.
CSXT	6811-HUNTINGTON	WV	579.
CSXT	6808-KENOVA	WV	585.
CSXT	6807-CATLETTSBURG	KY	587.
CSXT	6809-ASHLAND	KY	593.
CSXT	6846-SILOAM	KY	616.
CSXT	3162-CHILLICOTHE	OH	679.
CSXT	3095-COLUMBUS (BROAD	OH	721.
CSXT	3402-MARION	OH	766.
CSXT	3002-FOSTORIA	OH	809.
CSXT	3484-DESHLER	OH	837.
CSXT	3993-WELLSBORO	IN	985.
CSXT	4070-GARY	IN	1019.
CSXT	4073-CLARKE	IN	1023.
CSXT	4074-INDIANA HARBOR	IN	1026.
CSXT	4232-SOUTH CHICAGO	IL	1034.
CSXT	4231-BURNSIDE	IL	1037.
CSXT	4217-CHICAGO	IL	1049.
CNW	4217-CHICAGO	IL	1049.
CNW	4234-PROVISO	IL	1064.
CNW	4311-DE KALB	IL	1106.
CNW	4324-NELSON	IL	1151.
CNW	10304-CLINTON	IA	1183.
CNW	10289-CEDAR RAPIDS	IA	1264.
CNW	10265-MARSHALLTOWN	IA	1331.
CNW	10246-NEVADA	IA	1358.
CNW	10271-AMES	IA	1369.
CNW	10176-MISSOURI VALLEY	IA	1502.
CNW	10198-CALIFORNIA JCT	IA	1508.
CNW	11340-FREMONT	NE	1536.
UP	11340-FREMONT	NE	1536.
UP	11406-GRAND ISLAND	NE	1645.
UP	11410-GIBBON	NE	1671.
UP	11352-NORTH PLATTE	NE	1749.
UP	11358-O FALLONS	NE	1798.
UP	13703-JULESBURG	CO	1866.
UP	13465-CHEYENNE	WY	2012.
UP	13462-LARAMIE	WY	2064.
UP	13494-GRANGER	WY	2340.
UP	13568-OGDEN	UT	2479.
UP	13595-SALT LAKE CITY	UT	2515.
UP	13630-LYNNDYL	UT	2627.
UP	14766-VALLEY	NV	2944.
USG	14766-VALLEY	NV	2944.
USG	16333-YUCCA MOUNTAIN	NV	3043.

Figure E1-7 Representative Truck Routes from Jacksonville, FL to Department of Energy Management Sites

From: JACKSONVILLE, FL
To : SRP, SC

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I95		ROSINVILLE	N	I26	I95	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: JACKSONVILLE, FL
To : ID NATL ENG LAB, ID

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I295		JACKSONVILLE	W	I10	I295	FL
I10		WINFIELD	W	I10	I75	FL
I75		MACON	S	I475	I75	GA
I475		SMARR	E	I475	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: JACKSONVILLE, FL
To : HANFORD, WA

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I295		JACKSONVILLE	W	I10	I295	FL
I10		WINFIELD	W	I10	I75	FL
I75		MACON	S	I475	I75	GA
I475		SMARR	E	I475	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: JACKSONVILLE, FL
To : K-25, TN

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I295		JACKSONVILLE	W	I10	I295	FL
I10		WINFIELD	W	I10	I75	FL
I75		MACON	S	I475	I75	GA
I475		SMARR	E	I475	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: JACKSONVILLE, FL
To : MERCURY, NV

Routing through:

		JACKSONVILLE				FL
U23		JACKSONVILLE	W	I95	U23	FL
I95		JACKSONVILLE	NW	I95	U1	FL
I95 #		JACKSONVILLE	N	I295	I95	FL
I295		JACKSONVILLE	W	I10	I295	FL
I10		WINFIELD	W	I10	I75	FL
I75		MACON	S	I475	I75	GA
I475		SMARR	E	I475	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

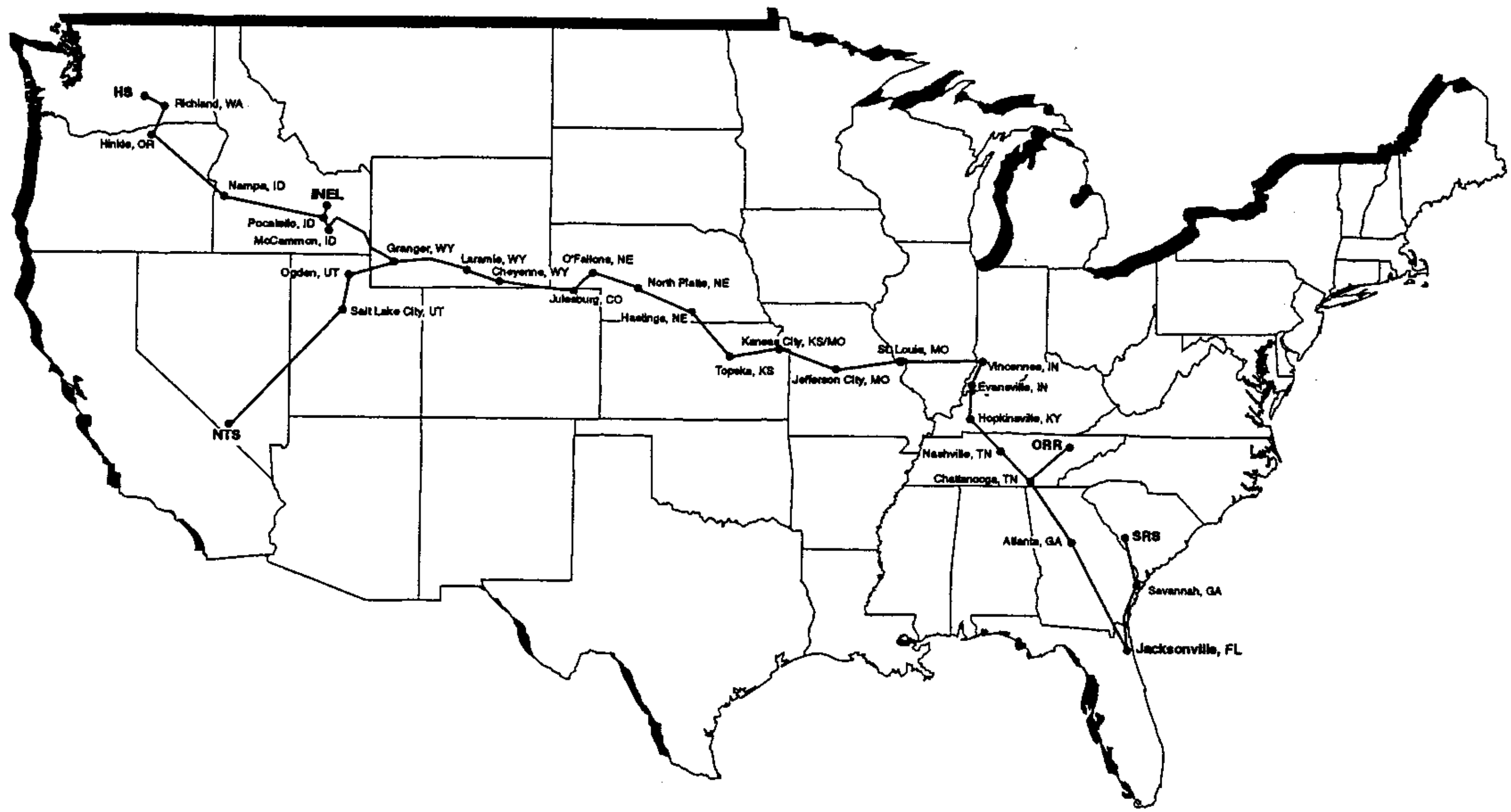


Figure E1-8 Representative Rail Routes from Jacksonville, FL to Department of Energy Management Sites

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8080-NAHUNTA	GA	64.
CSXT	8007-SAVANNAH	GA	145.
CSXT	7739-FAIRFAX	SC	213.
CSXT	7732-ROBBINS	SC	242.
CSXT	7717-DUNBARTON / WELL	SC	251.
USG	7717-DUNBARTON / WELL	SC	251.
USG	15359-SRP	SC	259.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13369-MC CAMMON	ID	2445.
UP	13370-POCATELLO	ID	2468.
UP	13336-SCOVILLE	ID	2524.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13369-MC CAMMON	ID	2445.
UP	13370-POCATELLO	ID	2468.
UP	13412-NAMPA	ID	2710.
UP	14220-PENDLETON	OR	2979.
UP	14223-HINKLE	OR	3010.
UP	13894-WALLULA	WA	3039.
UP	13964-KENNEWICK	WA	3054.
UP	13941-RICHLAND	WA	3062.
USG	13941-RICHLAND	WA	3062.
USG	16212-HANFORD S 300	WA	3070.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
NS	7235-CHATTANOOGA	TN	468.
NS	7260-HARRIMAN	TN	550.
NS	15316-K-25	TN	565.

ROUTE FROM: CSXT 8269-JACKSONVILLE, FL
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	8269-JACKSONVILLE	FL	0.
CSXT	8243-CALLAHAN	FL	17.
CSXT	8086-FOLKSTON	GA	39.
CSXT	8079-WAYCROSS	GA	71.
CSXT	8069-CORDELE	GA	179.
CSXT	8144-MANCHESTER	GA	260.
CSXT	7914-ATLANTA	GA	337.
CSXT	7907-MARIETTA	GA	347.
CSXT	7889-CARTERSVILLE	GA	379.
CSXT	7888-DALTON	GA	430.
CSXT	7235-CHATTANOOGA	TN	468.
CSXT	7187-TULLAHOMA	TN	549.
CSXT	7202-NASHVILLE	TN	628.
CSXT	7201-MADISON	TN	638.
CSXT	7061-HOPKINSVILLE	KY	698.
CSXT	3839-HENDERSON	KY	785.
CSXT	3838-EVANSVILLE	IN	798.
CSXT	3812-VINCENNES	IN	848.
CSXT	4952-SALEM	IL	927.
CSXT	10859-EAST ST LOUIS	IL	992.
<TR>	10859-EAST ST LOUIS	IL	992.
<TR>	10858-ST LOUIS	MO	998.
UP	10858-ST LOUIS	MO	998.
UP	10656-JEFFERSON CITY	MO	1120.
UP	10616-KANSAS CITY	MO	1296.
UP	10617-KANSAS CITY	KS	1299.
UP	11823-LAWRENCE	KS	1338.
UP	11697-TOPEKA	KS	1368.
UP	11696-MENOKEN	KS	1373.
UP	11681-MARYSVILLE	KS	1448.
UP	11405-HASTINGS	NE	1558.
UP	11410-GIBBON	NE	1584.
UP	11352-NORTH PLATTE	NE	1662.
UP	11358-O FALLONS	NE	1711.
UP	13703-JULESBURG	CO	1779.
UP	13465-CHEYENNE	WY	1925.
UP	13462-LARAMIE	WY	1977.
UP	13494-GRANGER	WY	2253.
UP	13568-OGDEN	UT	2392.
UP	13595-SALT LAKE CITY	UT	2427.
UP	13630-LYNNDYL	UT	2540.
UP	14766-VALLEY	NV	2857.
USG	14766-VALLEY	NV	2857.
USG	16333-YUCCA MOUNTAIN	NV	2956.

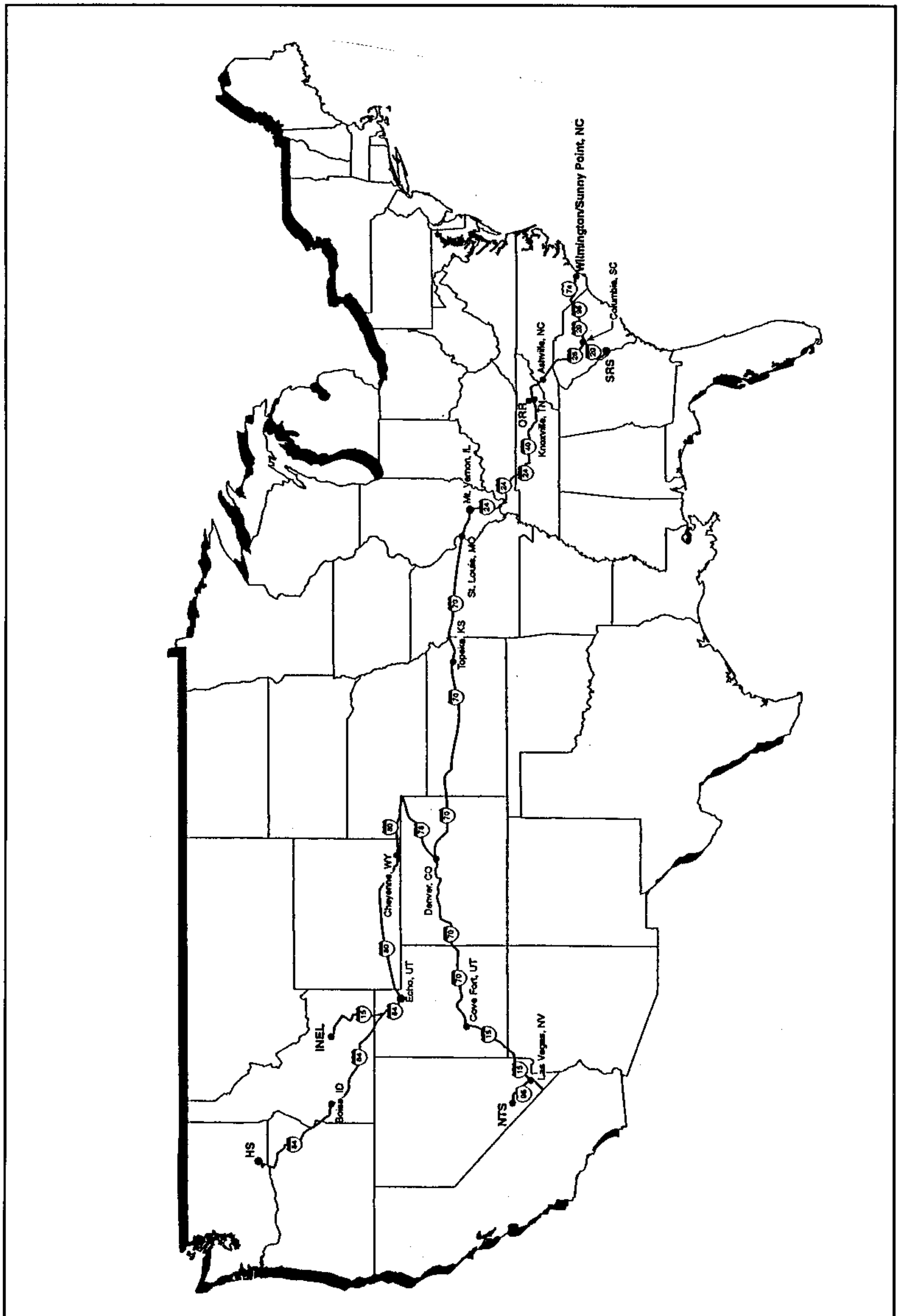


Figure E1-9 Representative Truck Routes from MOTSU, NC to Department of Energy Management Sites

From: SUNNY POINT TM, NC
To : SRP, SC

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		BENSON	NE	I40	I95	NC
I95		FLORENCE	W	I20	I95	SC
I20		MARTINEZ	E	I20	X65	GA
S28		BEECH ISLAND		U278	S125	SC
S125		SRP BARCD 1		S125	LC	SC

From: SUNNY POINT TM, NC
To : ID NATL ENG LAB, ID

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB		U20	LOCL	ID

From: SUNNY POINT TM, NC
To : HANFORD, WA

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	X5	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SUNNY POINT TM, NC
To : K-25, TN

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		KINGSTON	E	I40	X356	TN
S58		K-25				TN

From: SUNNY POINT TM, NC
To : MERCURY, NV

Routing through:

		SUNNY POINT TM				NC
LOCAL		SUNNY POINT	N	S133	LOCL	NC
S133		LELAND	SE	U17	S133	NC
U17	U74	WILMINGTON	W	U17	U421	NC
U17	U421	WILMINGTON		U117	U17	NC
U17	U74	WILMINGTON	E	U17	S132	NC
S132		WILMINGTON	NE	I40	X420	NC
I40		RALEIGH	SE	I40	I440	NC
I40	I440	RALEIGH	SW	I40	I440	NC
I40		HILLSBOROUGH	S	I40	I85	NC
I40	I85	GREENSBORO	S	I40	I85	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	FARRAGUT	W	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24		NASHVILLE	SE	I24	I440	TN
I440		NASHVILLE	W	I40	I440	TN
I40		NASHVILLE	W	I265	I40	TN
I265		NASHVILLE	N	I24	I265	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70		BONNER SPRINGS	N	I70	X224	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		LAS VEGAS	W	U95	U95B	NV
U95BU		LAS VEGAS	NW	U95	U95B	NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

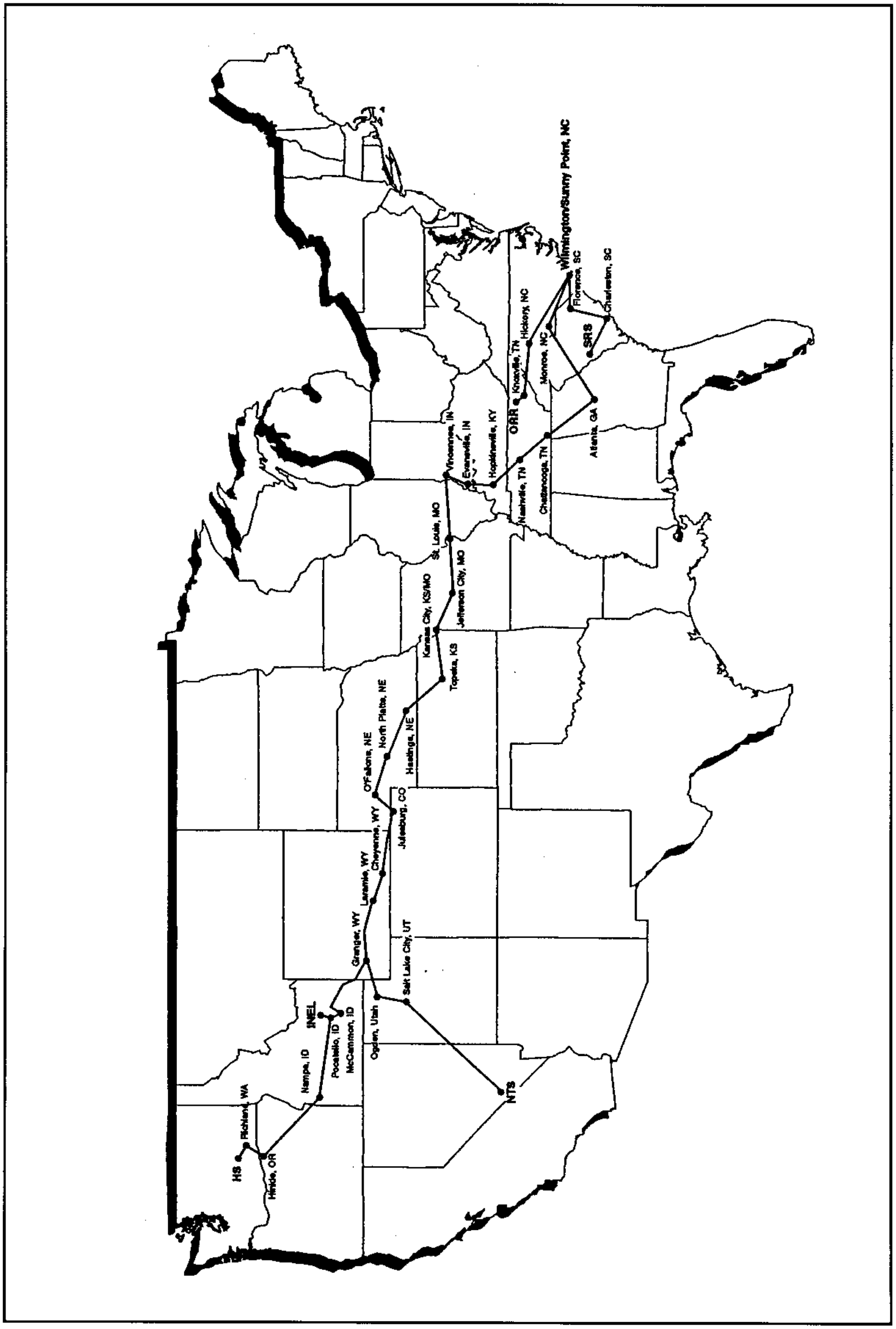


Figure E1-10 Representative Rail Routes from MOTSU, NC to Department of Energy Management Sites

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7671-DILLON	SC	115.
CSXT	7675-FLORENCE	SC	144.
CSXT	7690-CHARLESTON	SC	242.
CSXT	7739-FAIRFAX	SC	336.
CSXT	7732-ROBBINS	SC	365.
CSXT	7717-DUNBARTON / WELL	SC	374.
USG	7717-DUNBARTON / WELL	SC	374.
USG	15359-SRP	SC	382.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7834-CLINTON	SC	271.
CSXT	7838-GREENWOOD	SC	299.
CSXT	7956-ATHENS	GA	380.
CSXT	17420-ATLANTA BELT JCT	GA	442.
CSXT	17424-TILFORD YARD (S.	GA	450.
CSXT	7907-MARIETTA	GA	465.
CSXT	7889-CARTERSVILLE	GA	493.
CSXT	7888-DALTON	GA	544.
CSXT	7235-CHATTANOOGA	TN	581.
CSXT	7224-WAUHATCHIE	TN	587.
CSXT	7187-TULLAHOMA	TN	663.
CSXT	7201-MADISON	TN	751.
CSXT	7061-HOPKINSVILLE	KY	811.
CSXT	3839-HENDERSON	KY	876.
CSXT	3838-EVANSVILLE	IN	889.
CSXT	3812-VINCENNES	IN	939.
CSXT	4952-SALEM	IL	1017.
CSXT	10859-EAST ST LOUIS	IL	1084.
TRRA	10859-EAST ST LOUIS	IL	1084.
UP	10859-EAST ST LOUIS	IL	1084.
UP	10858-ST LOUIS	MO	1085.
UP	10860-PACIFIC	MO	1112.
UP	10656-JEFFERSON CITY	MO	1210.
UP	10616-KANSAS CITY	MO	1387.
UP	10617-KANSAS CITY	KS	1389.
UP	11823-LAWRENCE	KS	1428.
UP	11697-TOPEKA	KS	1458.
UP	11696-MENOKEN	KS	1463.
UP	11681-MARYSVILLE	KS	1538.
UP	11405-HASTINGS	NE	1646.
UP	11410-GIBBON	NE	1672.
UP	11352-NORTH PLATTE	NE	1791.
UP	11358-O FALLONS	NE	1802.
UP	13703-JULESBURG	CO	1870.
UP	13465-CHEYENNE	WY	2016.
UP	13462-LARAMIE	WY	2068.
UP	13494-GRANGER	WY	2344.
UP	13369-MC CAMMON	ID	2536.
UP	13370-POCATELLO	ID	2559.
UP	13336-SCOVILLE	ID	2615.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7406-CHARLOTTE	NC	202.
CSXT	7395-MOUNT HOLLY	NC	213.
CSXT	7388-BOSTIC	NC	276.
CSXT	7387-MARION	NC	303.
CSXT	7311-JOHNSON CITY	TN	407.
CSXT	7312-KINGSPORT	TN	438.
CSXT	7313-FRISCO	TN	444.
CSXT	6946-BEAVER JCT	KY	570.
CSXT	6807-CATLETTSBURG	KY	654.
CSXT	6809-ASHLAND	KY	660.
CSXT	6846-SILOAM	KY	684.
CSXT	3162-CHILLICOTHE	OH	746.
CSXT	3095-COLUMBUS (BROAD	OH	788.
CSXT	3402-MARION	OH	833.
CSXT	3002-FOSTORIA	OH	876.
CSXT	3484-DESHLER	OH	904.
CSXT	3993-WELLSBORO	IN	1053.
CSXT	4070-GARY	IN	1087.
CSXT	4073-CLARKE	IN	1091.
CSXT	4076-HAMMOND	IN	1097.
CSXT	4228-BURNHAM / CALUME	IL	1099.
CSXT	4163-BLUE ISLAND	IL	1107.
IHB	4163-BLUE ISLAND	IL	1107.
IHB	4172-ARGO	IL	1119.
IHB	4170-LA GRANGE	IL	1123.
BN	4170-LA GRANGE	IL	1123.
BN	4190-AURORA	IL	1148.
BN	4317-SAVANNA	IL	1239.
BN	4327-EAST DUBUQUE	IL	1279.
BN	5736-LA CROSSE	WI	1390.
BN	9830-ST PAUL	MN	1511.
BN	9793-SOO LINE JCT	MN	1514.
BN	15603-EAST MINNEAPOLIS	MN	1521.
BN	9798-NORTHTOWN	MN	1527.
BN	9826-COON CREEK	MN	1532.
BN	9671-SAUK RAPIDS	MN	1582.
BN	9663-STAPLES	MN	1647.
BN	11131-MOORHEAD	MN	1761.
BN	11132-FARGO	ND	1764.
BN	11134-CASSELTON	ND	1784.
BN	10935-SURREY	ND	1999.
BN	10936-MINOT	ND	2005.
BN	15740-WILLISTON	ND	2117.
BN	13168-HAVRE	MT	2436.
BN	13089-SHELBY	MT	2537.
BN	13300-SANDPOINT	ID	2874.
BN	13828-SPOKANE	WA	2937.
BN	13890-PASCO	WA	3088.
BN	13964-KENNEWICK	WA	3090.
WCRC	13964-KENNEWICK	WA	3090.
UP	13964-KENNEWICK	WA	3090.
UP	13941-RICHLAND	WA	3098.
USG	13941-RICHLAND	WA	3098.
USG	16212-HANFORD S 300	WA	3106.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7406-CHARLOTTE	NC	202.
CSXT	7395-MOUNT HOLLY	NC	213.
CSXT	7388-BOSTIC	NC	276.
CSXT	7387-MARION	NC	303.
NS	7387-MARION	NC	303.
NS	7343-ASHEVILLE	NC	343.
NS	7318-MORRISTOWN	TN	423.
NS	16403-SEVIER YARD W EN	TN	456.
NS	16401-COSTER	TN	464.
NS	7288-DOSSETT	TN	488.
NS	15316-K-25	TN	509.

ROUTE FROM: USG 7611-SUNNY POINT JCT, NC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
USG	7611-SUNNY POINT JCT	NC	0.
CSXT	7611-SUNNY POINT JCT	NC	0.
CSXT	7625-WILMINGTON	NC	9.
CSXT	7620-PEMBROKE	NC	95.
CSXT	7470-HAMLET	NC	126.
CSXT	7407-MONROE	NC	179.
CSXT	7834-CLINTON	SC	271.
CSXT	7838-GREENWOOD	SC	299.
CSXT	7956-ATHENS	GA	380.
CSXT	17420-ATLANTA BELT JCT	GA	442.
CSXT	17424-TILFORD YARD (S.	GA	450.
CSXT	7907-MARIETTA	GA	465.
CSXT	7889-CARTERSVILLE	GA	493.
CSXT	7888-DALTON	GA	544.
CSXT	7235-CHATTANOOGA	TN	581.
CSXT	7224-WAUHATCHIE	TN	587.
CSXT	7187-TULLAHOMA	TN	663.
CSXT	7201-MADISON	TN	751.
CSXT	7061-HOPKINSVILLE	KY	811.
CSXT	3839-HENDERSON	KY	876.
CSXT	3838-EVANSVILLE	IN	889.
CSXT	3812-VINCENNES	IN	939.
CSXT	4952-SALEM	IL	1017.
CSXT	10859-EAST ST LOUIS	IL	1084.
TRRA	10859-EAST ST LOUIS	IL	1084.
UP	10859-EAST ST LOUIS	IL	1084.
UP	10858-ST LOUIS	MO	1085.
UP	10860-PACIFIC	MO	1112.
UP	10656-JEFFERSON CITY	MO	1210.
UP	10616-KANSAS CITY	MO	1387.
UP	10617-KANSAS CITY	KS	1389.
UP	11823-LAWRENCE	KS	1428.
UP	11697-TOPEKA	KS	1458.
UP	11696-MENOKEN	KS	1463.
UP	11681-MARYSVILLE	KS	1538.
UP	11405-HASTINGS	NE	1646.
UP	11410-GIBBON	NE	1672.
UP	11352-NORTH PLATTE	NE	1791.
UP	11358-O FALLONS	NE	1802.
UP	13703-JULESBURG	CO	1870.
UP	13465-CHEYENNE	WY	2016.
UP	13462-LARAMIE	WY	2068.
UP	13494-GRANGER	WY	2344.
UP	13568-OGDEN	UT	2487.
UP	13595-SALT LAKE CITY	UT	2523.
UP	13630-LYNNDYL	UT	2635.
UP	14766-VALLEY	NV	2952.
USG	14766-VALLEY	NV	2952.
USG	16333-YUCCA MOUNTAIN	NV	3051.

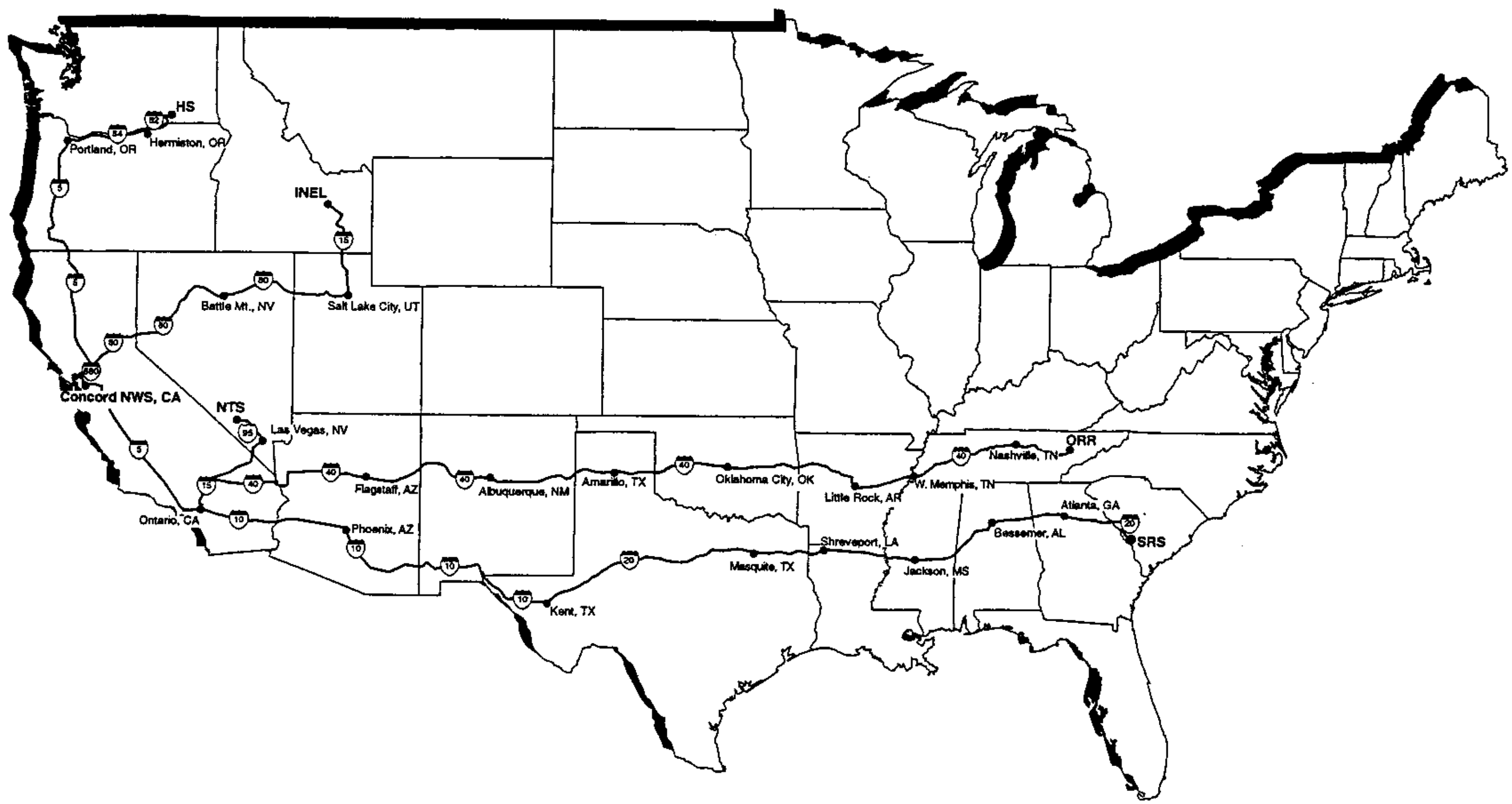


Figure E1-11 Representative Truck Routes from Naval Weapons Station, Concord, CA to Department of Energy Management Sites

From: CONCORD NW I680 S4 CA
To : SRL, SC

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		PHOENIX	W	I10	I17	AZ
I17		PHOENIX	SE	I10	I17	AZ
I10		KENT	E	I10	I20	TX
I20		DUNCANVILLE	NE	I20	U67	TX
I20	I635	MESQUITE	NW	I20	I635	TX
I20		SHREVEPORT	W	I20	I220	LA
I220		BOSSIER CITY	E	I20	I220	LA
I20		JACKSON	SW	I20	I55	MS
I20	I55	JACKSON	S	I20	I55	MS
I20		MERIDIAN	W	I20	I59	MS
I20	I59	BESSEMER	SW	I20	I459	AL
I459		IRONDALE	E	I20	I459	AL
I20		ATLANTA	W	I20	I285	GA
I285		RED OAK	E	I285	I85	GA
I285	I85	COLLEGE PARK	S	I285	I85	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: CONCORD NW I680 S4 CA
To : ID NATL ENG LAB, ID

Routing through:

		CONCORD	NW	I680	S4	CA
I680		MARTINEZ	E	I680	LOCL	CA
I680#		BENICIA	E	I680	I780	CA
I680		ROCKVILLE	SW	I680	I80	CA
I80		SALT LAKE CITY	W	I215	I80	UT
I215		N SALT LAKE		I15	I215	UT
I15		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: CONCORD NW I680 S4 CA
To : HANFORD, WA

Routing through:

		CONCORD	NW	I680	S4	CA
I680		MARTINEZ	E	I680	LOCL	CA
I680#		BENICIA	E	I680	I780	CA
I680		ROCKVILLE	SW	I680	I80	CA
I80		VACAVILLE	E	I505	I80	CA
I505		DUNNIGAN	S	I5	I505	CA
I5		TUALATIN	S	I205	I5	OR
I205		PORTLAND	E	I205	I84	OR
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: CONCORD NW I680 S4 CA
To : K-25, TN

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		ONTARIO	E	I10	I15	CA
I15		BARSTOW		I15	I40	CA
I40		OKLAHOMA CITY	W	I40	I44	OK
I44		OKLAHOMA CITY	SW	I240	I44	OK
I240		OKLAHOMA CITY	SE	I240	I40	OK
I40		WEST MEMPHIS	N	I40	I55	AR
I40	I55	WEST MEMPHIS	E	I40	I55	AR
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: CONCORD NW I680 S4 CA
To : MERCURY, NV

Routing through:

		CONCORD	NW	I680	S4	CA
I680		DUBLIN	SE	I580	I680	CA
I580		VERNALIS	W	I5	I580	CA
I5		SAN FERNANDO	NW	I210	I5	CA
I210		POMONA	NW	I10	I210	CA
I10		ONTARIO	E	I10	I15	CA
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

Figure E1-12 Representative Rail Routes from Naval Weapons Stations, Concord, CA to Department of Energy Management Sites

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14470-PITTSBURG	CA	14.
SP	14498-LATHRUP	CA	79.
SP	14529-MODESTO	CA	106.
SP	14570-FRESNO	CA	204.
SP	14622-BAKERSFIELD	CA	309.
SP	14621-MOJAVE	CA	374.
SP	14692-PALMDALE	CA	408.
SP	14666-SAN BERNARDINO	CA	485.
SP	12971-YUMA	AZ	679.
SP	12972-WELLTON	AZ	720.
SP	12938-PICACHO	AZ	889.
SP	12942-TUCSON	AZ	938.
SP	12877-EL PASO	TX	1248.
SP	12611-SAN ANTONIO	TX	1866.
SP	12538-FLATONIA	TX	1958.
SP	12536-EAGLE LAKE	TX	2009.
SP	12399-HOUSTON	TX	2078.
SP	12341-BEAUMONT	TX	2165.
SP	12348-ORANGE	TX	2188.
SP	9124-LAKE CHARLES	LA	2231.
SP	9113-LAFAYETTE	LA	2297.
SP	9038-THIBODAUX JCT	LA	2394.
SP	8985-NEW ORLEANS	LA	2454.
CSXT	8985-NEW ORLEANS	LA	2454.
CSXT	8966-GULFPORT	MS	2528.
CSXT	8926-BILOXI	MS	2545.
CSXT	8967-PASCAGOULA	MS	2561.
CSXT	8597-MOBILE	AL	2600.
CSXT	8566-FLOMATON	AL	2653.
CSXT	8657-MONTGOMERY	AL	2777.
CSXT	8683-OPELIKA	AL	2843.
CSXT	8142-LA GRANGE	GA	2886.
CSXT	7914-ATLANTA	GA	2962.
CSXT	7961-AUGUSTA	GA	3137.
CSXT	7732-ROBBINS	SC	3166.
CSXT	7717-DUNBARTON / WELL	SC	3175.
USG	7717-DUNBARTON / WELL	SC	3175.
USG	15359-SRP	SC	3183.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14821-RENO	NV	194.
SP	14816-SPARKS	NV	209.
SP	14812-HAZEN	NV	242.
SP	14813-WINNEMUCCA	NV	376.
SP	14793-ELKO	NV	500.
SP	13568-OGDEN	UT	723.
UP	13568-OGDEN	UT	723.
UP	13369-MC CAMMON	ID	836.
UP	13370-POCATELLO	ID	859.
UP	13336-SCOVILLE	ID	915.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14383-MARYSVILLE	CA	108.
SP	14385-CHICO	CA	145.
SP	14387-TEHAMA	CA	177.
SP	14313-REDDING	CA	224.
SP	14250-KLAMATH FALLS	OR	386.
SP	14271-EUGENE	OR	579.
SP	14287-ALBANY	OR	623.
SP	14141-SALEM	OR	651.
SP	14179-PORTLAND	OR	704.
UP	14179-PORTLAND	OR	704.
UP	14197-OREGON TRUNK JCT	OR	799.
UP	14223-HINKLE	OR	891.
UP	13894-WALLULA	WA	920.
UP	13964-KENNEWICK	WA	935.
UP	13941-RICHLAND	WA	944.
USG	13941-RICHLAND	WA	944.
USG	16212-HANFORD S 300	WA	951.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14431-SUISUN-FAIRFIELD	CA	21.
SP	14409-DAVIS	CA	48.
SP	14411-SACRAMENTO	CA	62.
SP	14415-ROSEVILLE	CA	77.
SP	14821-RENO	NV	194.
SP	14816-SPARKS	NV	209.
SP	14812-HAZEN	NV	242.
SP	14813-WINNEMUCCA	NV	376.
SP	14793-ELKO	NV	500.
SP	13568-OGDEN	UT	723.
SP	13595-SALT LAKE CITY	UT	758.
SP	13610-PROVO	UT	803.
SP	13611-SPRINGVILLE	UT	808.
SP	13613-THISTLE	UT	822.
SP	13646-GRAND JCT	CO	1048.
SP	13645-GLENWOOD SPRINGS	CO	1138.
SP	13673-DOTSERO	CO	1153.
SP	13764-PUEBLO	CO	1377.
SP	11673-HERINGTON	KS	1837.
SP	11697-TOPEKA	KS	1914.
SP	11823-LAWRENCE	KS	1944.
SP	10617-KANSAS CITY	KS	1982.
SP	10616-KANSAS CITY	MO	1985.
SP	10627-PLEASANT HILL	MO	2017.
SP	10656-JEFFERSON CITY	MO	2133.
SP	10858-ST LOUIS	MO	2255.
SP	10859-EAST ST LOUIS	IL	2261.
<TR>	10859-EAST ST LOUIS	IL	2261.
NS	10859-EAST ST LOUIS	IL	2261.
NS	4953-CENTRALIA	IL	2319.
NS	4954-MOUNT VERNON	IL	2341.
NS	4797-MOUNT CARMEL	IL	2404.
NS	7009-JEFFERSONVILLE	IN	2529.
NS	7008-LOUISVILLE	KY	2533.
NS	6979-DANVILLE	KY	2633.
NS	7260-HARRIMAN	TN	2795.
NS	15316-K-25	TN	2810.

ROUTE FROM: SP 14467-MARTINEZ, CA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
SP	14467-MARTINEZ	CA	0.
SP	14470-PITTSBURG	CA	14.
SP	14498-LATHRUP	CA	79.
SP	14529-MODESTO	CA	106.
SP	14570-FRESNO	CA	204.
SP	14622-BAKERSFIELD	CA	309.
SP	14621-MOJAVE	CA	374.
SP	14692-PALMDALE	CA	408.
SP	14666-SAN BERNARDINO	CA	485.
UP	14666-SAN BERNARDINO	CA	485.
UP	14664-BARSTOW	CA	563.
UP	14762-LAS VEGAS	NV	736.
UP	14766-VALLEY	NV	751.
USG	14766-VALLEY	NV	751.
USG	16333-YUCCA MOUNTAIN	NV	850.

Figure E1-13 Representative Truck Routes from Portland, OR to Department of Energy Management Sites

From: PORTLAND, OR
To : SRL, SC

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: PORTLAND, OR
To : ID NATL ENG LAB, ID

Routing through:

		PORTLAND				OR
I84		RAFT RIVER	W	I84	I86	ID
I86		CHUBBUCK	E	I15	I86	ID
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: PORTLAND, OR
To : HANFORD, WA

Routing through:

		PORTLAND				OR
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: PORTLAND, OR
To : K-25, TN

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: PORTLAND, OR
To : MERCURY, NV

Routing through:

		PORTLAND				OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

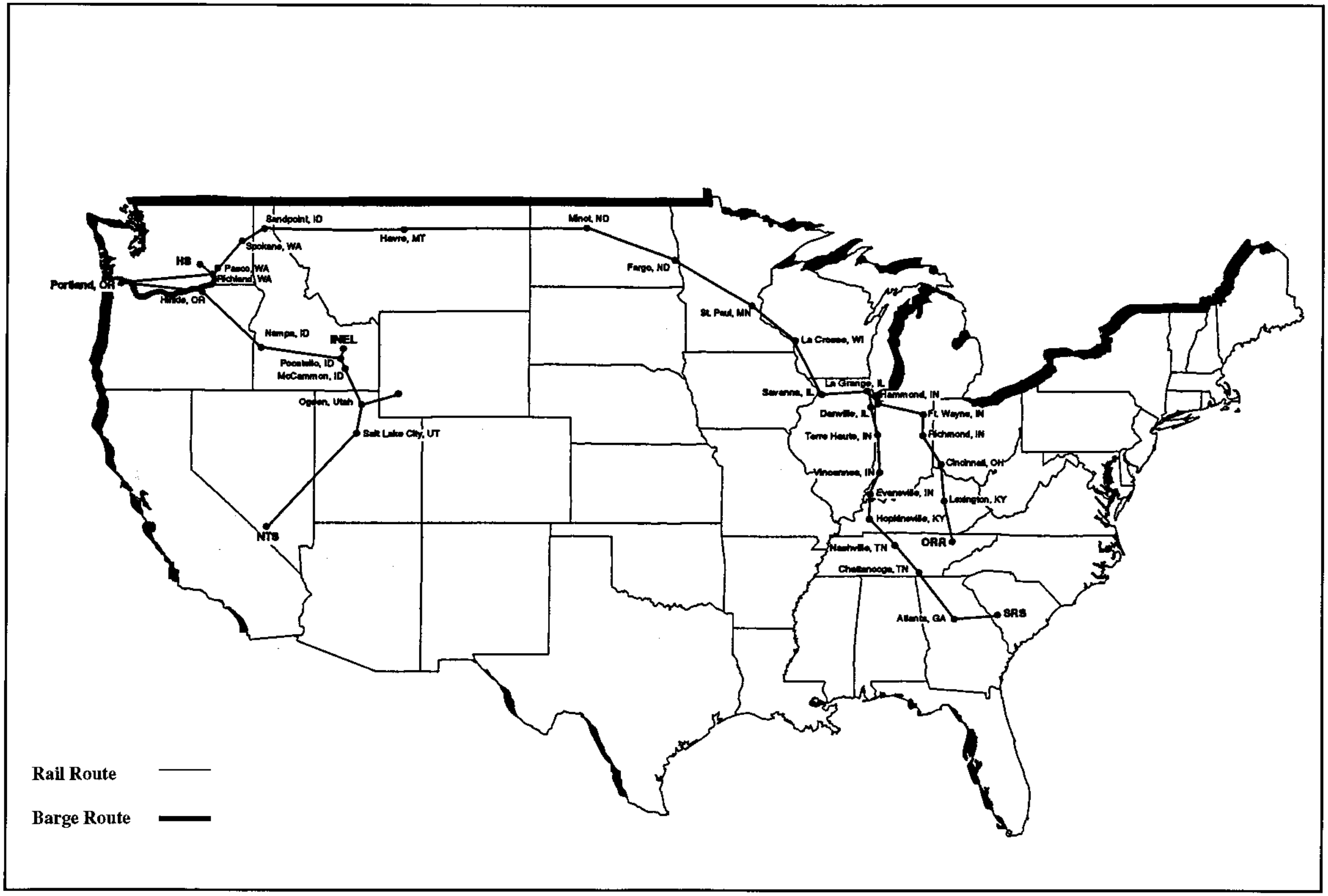


Figure E1-14 Representative Rail and Barge Routes from Portland, OR to Department of Energy Management Sites

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
BN	13828-SPOKANE	WA	373.
BN	13300-SANDPOINT	ID	436.
BN	13089-SHELBY	MT	773.
BN	13168-HAVRE	MT	874.
BN	15740-WILLISTON	ND	1193.
BN	10936-MINOT	ND	1305.
BN	10935-SURREY	ND	1311.
BN	11134-CASSELTON	ND	1526.
BN	11132-FARGO	ND	1546.
BN	11131-MOORHEAD	MN	1549.
BN	9663-STAPLES	MN	1663.
BN	9671-SAUK RAPIDS	MN	1728.
BN	9826-COON CREEK	MN	1778.
BN	9798-NORTHTOWN	MN	1787.
BN	9830-ST PAUL	MN	1799.
BN	5736-LA CROSSE	WI	1920.
BN	4327-EAST DUBUQUE	IL	2031.
BN	4317-SAVANNA	IL	2071.
BN	4190-AURORA	IL	2162.
BN	4170-LA GRANGE	IL	2187.
IHB	4170-LA GRANGE	IL	2187.
IHB	4172-ARGO	IL	2191.
IHB	4163-BLUE ISLAND	IL	2203.
IHB	4223-DOLTON / RIVERDAIL		2207.
CSXT	4223-DOLTON / RIVERDAIL		2207.
CSXT	4206-CHICAGO HEIGHTS	IL	2217.
CSXT	4636-WATSEKA	IL	2268.
CSXT	4642-DANVILLE	IL	2313.
CSXT	3863-TERRE HAUTE	IN	2370.
CSXT	3812-VINCENNES	IN	2423.
CSXT	3838-EVANSVILLE	IN	2473.
CSXT	3839-HENDERSON	KY	2486.
CSXT	7061-HOPKINSVILLE	KY	2573.
CSXT	7201-MADISON	TN	2634.
CSXT	7202-NASHVILLE	TN	2644.
CSXT	7187-TULLAHOMA	TN	2723.
CSXT	7235-CHATTANOOGA	TN	2803.
CSXT	7888-DALTON	GA	2841.
CSXT	7889-CARTERSVILLE	GA	2892.
CSXT	7907-MARIETTA	GA	2924.
CSXT	7914-ATLANTA	GA	2934.
CSXT	7961-AUGUSTA	GA	3109.
CSXT	7732-ROBBINS	SC	3138.
CSXT	7717-DUNBARTON / WELL	SC	3147.
USG	7717-DUNBARTON / WELL	SC	3147.
USG	15359-SRP	SC	3155.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
UP	14179-PORTLAND	OR	0.
UP	14197-OREGON TRUNK JCT	OR	95.
UP	14223-HINKLE	OR	187.
UP	14220-PENDLETON	OR	218.
UP	13412-NAMPA	ID	487.
UP	13370-POCATELLO	ID	729.
UP	13336-SCOVILLE	ID	785.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
WCRC	13890-PASCO	WA	222.
WCRC	13964-KENNEWICK	WA	223.
WCRC	13941-RICHLAND	WA	231.
USG	13941-RICHLAND	WA	231.
USG	16212-HANFORD S 300	WA	239.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
BN	14180-VANCOUVER	WA	7.
BN	13964-KENNEWICK	WA	220.
BN	13890-PASCO	WA	222.
BN	13828-SPOKANE	WA	373.
BN	13300-SANDPOINT	ID	436.
BN	13089-SHELBY	MT	773.
BN	13168-HAVRE	MT	874.
BN	15740-WILLISTON	ND	1193.
BN	10936-MINOT	ND	1305.
BN	10935-SURREY	ND	1311.
BN	11134-CASSELTON	ND	1526.
BN	11132-FARGO	ND	1546.
BN	11131-MOORHEAD	MN	1549.
BN	9663-STAPLES	MN	1663.
BN	9671-SAUK RAPIDS	MN	1728.
BN	9826-COON CREEK	MN	1778.
BN	9798-NORTHTOWN	MN	1787.
BN	9830-ST PAUL	MN	1799.
BN	5736-LA CROSSE	WI	1920.
BN	4327-EAST DUBUQUE	IL	2031.
BN	4317-SAVANNA	IL	2071.
BN	4190-AURORA	IL	2162.
BN	4170-LA GRANGE	IL	2187.
IHB	4170-LA GRANGE	IL	2187.
IHB	4172-ARGO	IL	2191.
IHB	4163-BLUE ISLAND	IL	2203.
IHB	4228-BURNHAM / CALUME	IL	2211.
NS	4228-BURNHAM / CALUME	IL	2211.
NS	4076-HAMMOND	IN	2213.
NS	4064-HOBART	IN	2229.
NS	4020-ARGOS	IN	2292.
NS	3548-FORT WAYNE	IN	2351.
NS	3650-MUNCIE	IN	2415.
NS	3688-RICHMOND	IN	2460.
NS	3251-HAMILTON	OH	2515.
NS	3234-IVORYDALE	OH	2532.
NS	3228-CINCINNATI	OH	2539.
NS	6850-LEXINGTON	KY	2613.
NS	6979-DANVILLE	KY	2650.
NS	7260-HARRIMAN	TN	2812.
NS	15316-K-25	TN	2827.

ROUTE FROM: BN 14179-PORTLAND, OR
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	14179-PORTLAND	OR	0.
UP	14179-PORTLAND	OR	0.
UP	14197-OREGON TRUNK JCT	OR	95.
UP	14223-HINKLE	OR	187.
UP	14220-PENDLETON	OR	218.
UP	13412-NAMPA	ID	487.
UP	13370-POCATELLO	ID	729.
UP	13369-MC CAMMON	ID	752.
UP	13568-OGDEN	UT	865.
UP	13595-SALT LAKE CITY	UT	901.
UP	13630-LYNNDYL	UT	1013.
UP	14766-VALLEY	NV	1330.
USG	14766-VALLEY	NV	1330.
USG	16333-YUCCA MOUNTAIN	NV	1429.

ROUTE FROM: BRG 16859-PORTLAND; PORT OF OR
TO: BRG 16851-RICHLAND; PORT OF WA

RR	NODE	STATE	DIST
BRG	16859-PORTLAND; PORT	OR	0.
BRG	16852-COLUMBIA/WILLAM	WA	10.
BRG	16853-VANCOUVER; PORT	WA	14.
BRG	16848-COLUMBIA/SNAKE	WA	231.
BRG	16851-RICHLAND; PORT	WA	243.

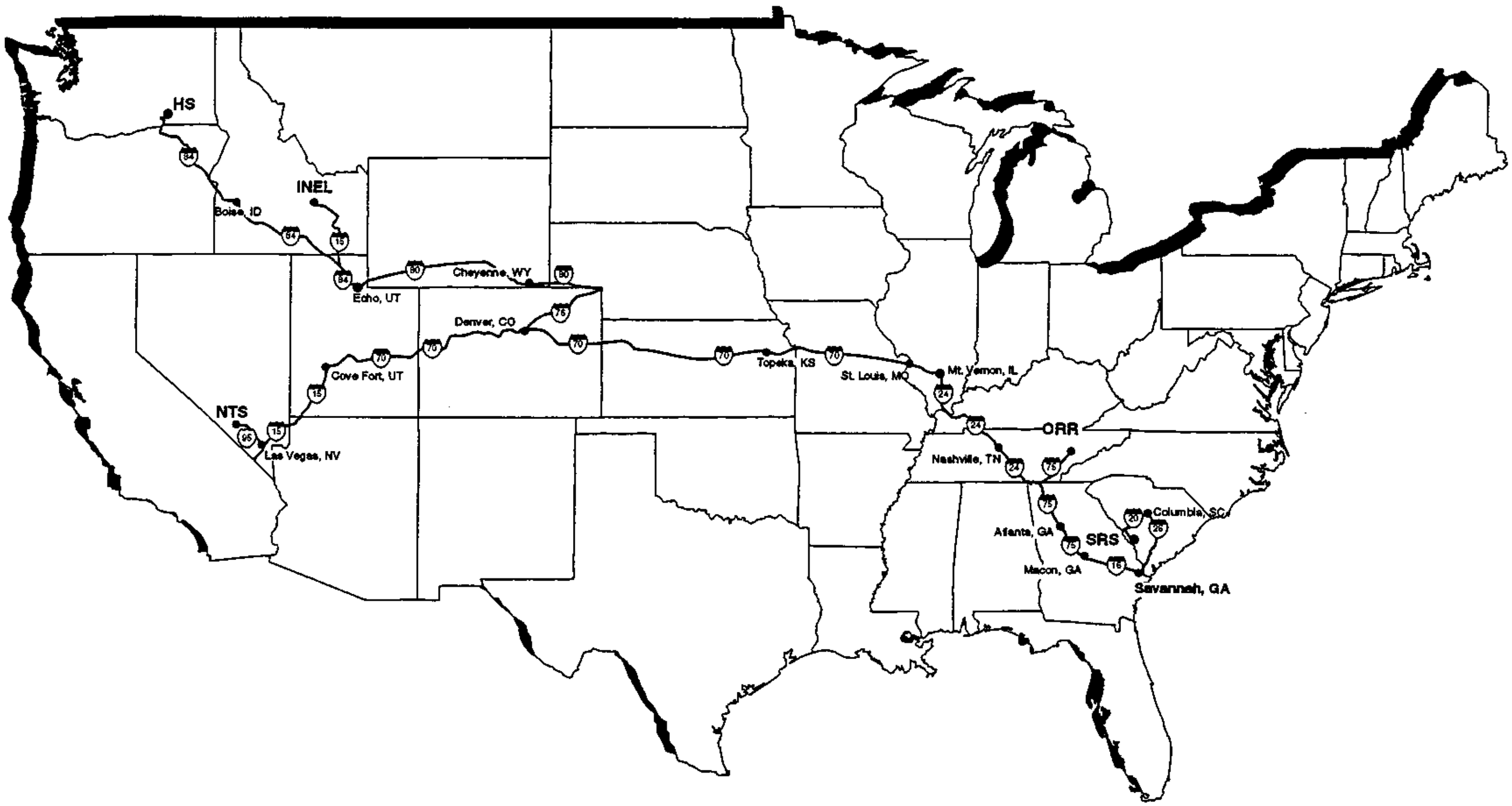


Figure E1-15 Representative Truck Routes from Savannah, GA to Department of Energy Management Sites

From: SAVANNAH, GA
To : SRP, SC

Routing through:

		SAVANNAH				GA
I16		POOLER	S	I16	I95	GA
I95		ROSINVILLE	N	I26	I95	SC
I26		COLUMBIA	NW	I20	I26	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: SAVANNAH, GA
To : ID NATL ENG LAB, ID

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: SAVANNAH, GA
To : HANFORD, WA

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SAVANNAH, GA
To : K-25, TN

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: SAVANNAH, GA
To : MERCURY, NV

Routing through:

		SAVANNAH				GA
I16		MACON	NW	I16	I75	GA
I75		HAPEVILLE	S	I285	I75	GA
I285		COLLEGE PARK	S	I285	I85	GA
I285	I85	RED OAK	E	I285	I85	GA
I285		ATLANTA	NW	I285	I75	GA
I75		EAST RIDGE	NE	I24	I75	TN
I24		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

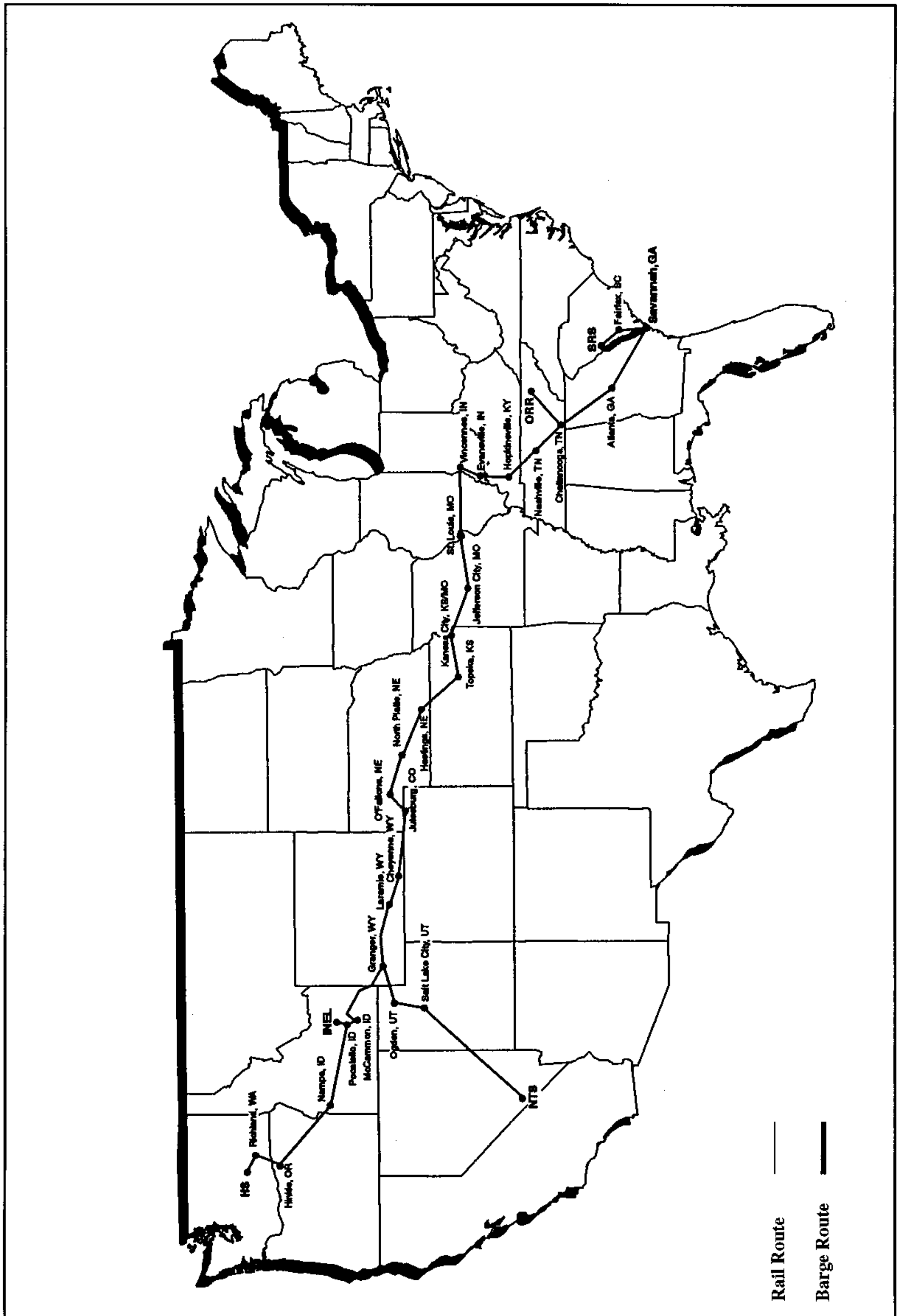


Figure E1-16 Representative Rail and Barge Routes from Savannah, GA to Department of Energy Management Sites

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	7739-FAIRFAX	SC	68.
CSXT	7732-ROBBINS	SC	97.
CSXT	7717-DUNBARTON / WELL	SC	106.
USG	7717-DUNBARTON / WELL	SC	106.
USG	15359-SRP	SC	114.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13369-MC CAMMON	ID	2467.
UP	13370-POCATELLO	ID	2490.
UP	13336-SCOVILLE	ID	2546.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13369-MC CAMMON	ID	2467.
UP	13370-POCATELLO	ID	2490.
UP	13412-NAMPA	ID	2732.
UP	14220-PENDLETON	OR	3000.
UP	14223-HINKLE	OR	3031.
UP	13894-WALLULA	WA	3060.
UP	13964-KENNEWICK	WA	3075.
UP	13941-RICHLAND	WA	3084.
USG	13941-RICHLAND	WA	3084.
USG	16212-HANFORD S 300	WA	3092.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
NS	7235-CHATTANOOGA	TN	490.
NS	7260-HARRIMAN	TN	572.
NS	15316-K-25	TN	587.

ROUTE FROM: CSXT 8007-SAVANNAH, GA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	8007-SAVANNAH	GA	0.
CSXT	8079-WAYCROSS	GA	93.
CSXT	8069-CORDELE	GA	201.
CSXT	8144-MANCHESTER	GA	282.
CSXT	7914-ATLANTA	GA	359.
CSXT	7907-MARIETTA	GA	369.
CSXT	7889-CARTERSVILLE	GA	401.
CSXT	7888-DALTON	GA	452.
CSXT	7235-CHATTANOOGA	TN	490.
CSXT	7187-TULLAHOMA	TN	571.
CSXT	7202-NASHVILLE	TN	650.
CSXT	7201-MADISON	TN	660.
CSXT	7061-HOPKINSVILLE	KY	720.
CSXT	3839-HENDERSON	KY	807.
CSXT	3838-EVANSVILLE	IN	820.
CSXT	3812-VINCENNES	IN	870.
CSXT	4952-SALEM	IL	949.
CSXT	10859-EAST ST LOUIS	IL	1014.
<TR>	10859-EAST ST LOUIS	IL	1014.
<TR>	10858-ST LOUIS	MO	1020.
UP	10858-ST LOUIS	MO	1020.
UP	10656-JEFFERSON CITY	MO	1142.
UP	10616-KANSAS CITY	MO	1318.
UP	10617-KANSAS CITY	KS	1321.
UP	11823-LAWRENCE	KS	1360.
UP	11697-TOPEKA	KS	1390.
UP	11696-MENOKEN	KS	1395.
UP	11681-MARYSVILLE	KS	1470.
UP	11405-HASTINGS	NE	1580.
UP	11410-GIBBON	NE	1606.
UP	11352-NORTH PLATTE	NE	1684.
UP	11358-O FALLONS	NE	1733.
UP	13703-JULESBURG	CO	1801.
UP	13465-CHEYENNE	WY	1947.
UP	13462-LARAMIE	WY	1999.
UP	13494-GRANGER	WY	2275.
UP	13568-OGDEN	UT	2414.
UP	13595-SALT LAKE CITY	UT	2449.
UP	13630-LYNNDYL	UT	2561.
UP	14766-VALLEY	NV	2878.
USG	14766-VALLEY	NV	2878.
USG	16333-YUCCA MOUNTAIN	NV	2977.

ROUTE FROM: BRG 16923-SAVANNAH RIV/ICWW, GA
TO: BRG 16932-SAVANNAH BLF L/D, GA

RR	NODE	STATE	DIST
BRG	16923-SAVANNAH RIV/ICW	GA	0.
BRG	16929-SAVANNAH PORT O	GA	10.
BRG	16932-SAVANNAH BLF L/D	GA	146.

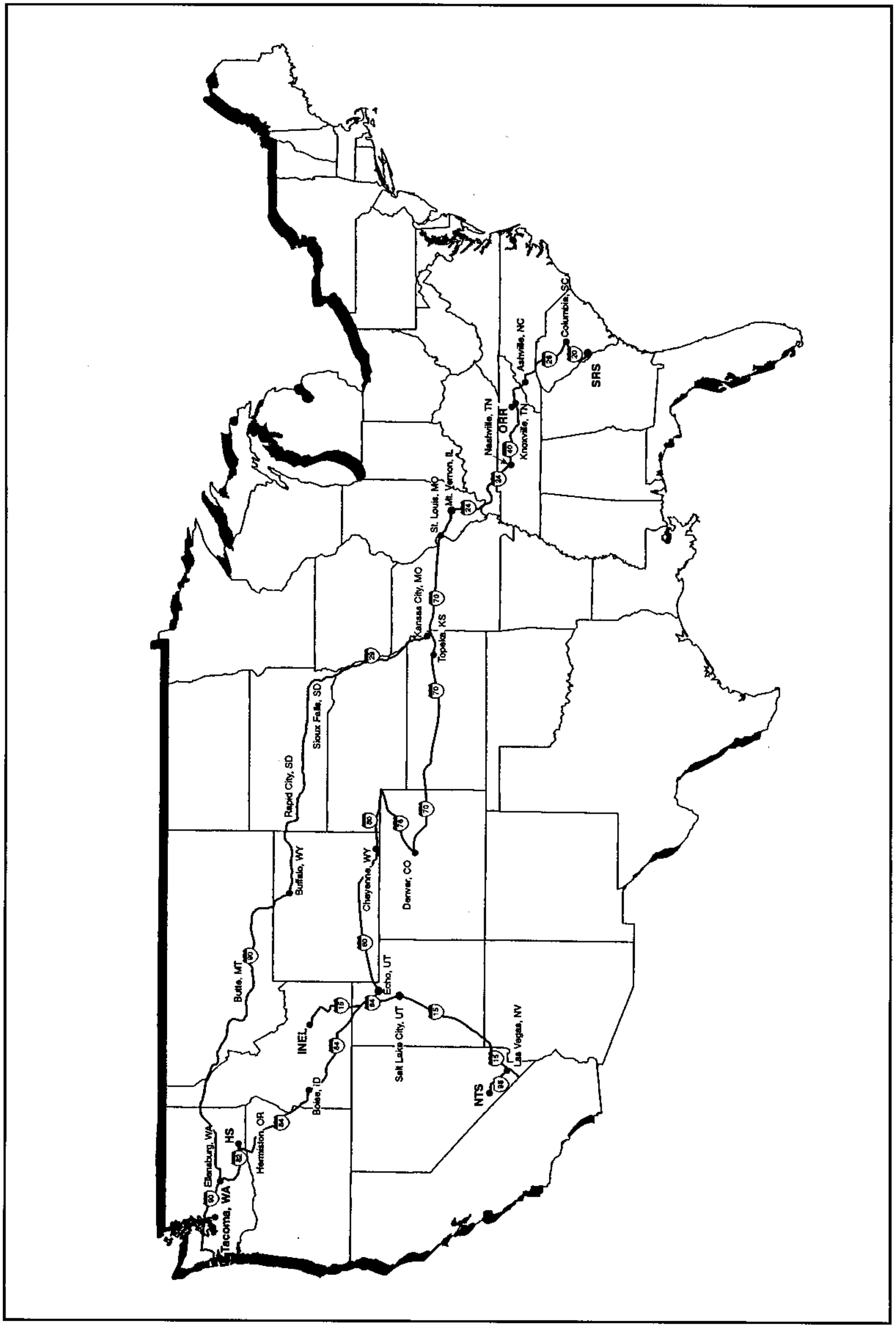


Figure E1-17 Representative Truck Routes from Tacoma, WA to Department of Energy Management Sites

From: TACOMA E I5 X135 WA
To : SRP, SC

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		BUTTE	W	I15	I90	MT
I15	I90	BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: TACOMA E I5 X135 WA
To : ID NATL ENG LAB, ID

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		HERMISTON	SW	I82	I84	OR
I84		RAFT RIVER	W	I84	I86	ID
I86		CHUBBUCK	E	I15	I86	ID
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: TACOMA E I5 X135 WA
To : HANFORD, WA

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: TACOMA E I5 X135 WA
To : K-25, TN

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		BUTTE	W	I15	I90	MT
I15	I90	BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: TACOMA E I5 X135 WA
To : MERCURY, NV

Routing through:

		TACOMA	E	I5	X135	WA
I5		RENTON	W	I405	I5	WA
I405		BELLEVUE	S	I405	I90	WA
I90		ELLENSBURG	SE	I82	I90	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

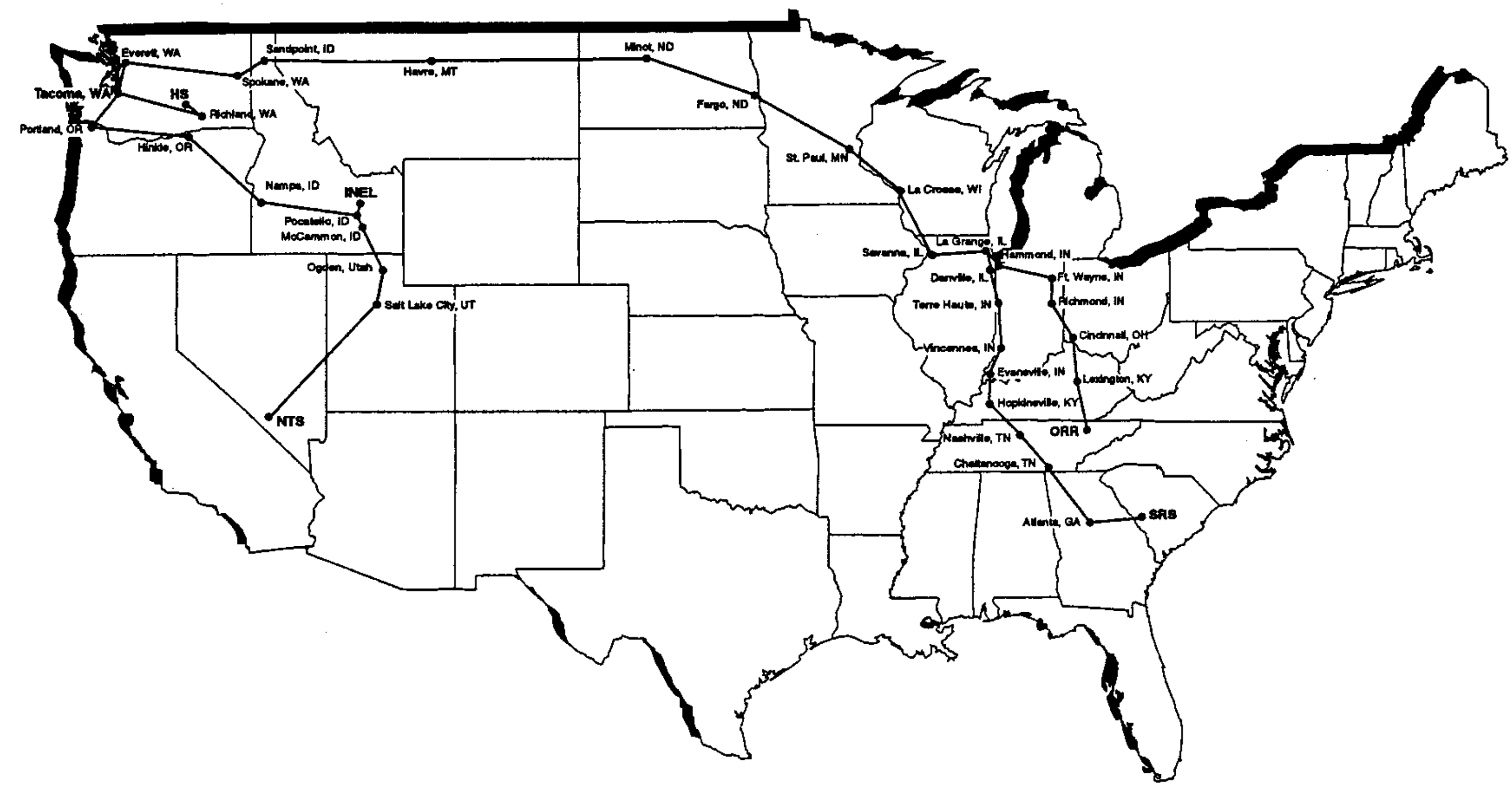


Figure E1-18 Representative Rail Routes from Tacoma, WA to Department of Energy Management Sites

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14011-SEATTLE	WA	42.
BN	14008-EVERETT	WA	74.
BN	13828-SPOKANE	WA	383.
BN	13300-SANDPOINT	ID	447.
BN	13089-SHELBY	MT	784.
BN	13168-HAVRE	MT	885.
BN	15740-WILLISTON	ND	1204.
BN	10936-MINOT	ND	1316.
BN	10935-SURREY	ND	1322.
BN	11134-CASSELTON	ND	1537.
BN	11132-FARGO	ND	1557.
BN	11131-MOORHEAD	MN	1560.
BN	9663-STAPLES	MN	1674.
BN	9671-SAUK RAPIDS	MN	1739.
BN	9826-COON CREEK	MN	1789.
BN	9798-NORTHTOWN	MN	1797.
BN	9830-ST PAUL	MN	1809.
BN	5736-LA CROSSE	WI	1930.
BN	4327-EAST DUBUQUE	IL	2041.
BN	4317-SAVANNA	IL	2081.
BN	4190-AURORA	IL	2172.
BN	4170-LA GRANGE	IL	2197.
- - -	- - -	- - -	- - -
IHB	4170-LA GRANGE	IL	2197.
IHB	4172-ARGO	IL	2201.
IHB	4163-BLUE ISLAND	IL	2213.
IHB	4223-DOLTON / RIVERDA	IL	2217.
- - -	- - -	- - -	- - -
CSXT	4223-DOLTON / RIVERDA	IL	2217.
CSXT	4206-CHICAGO HEIGHTS	IL	2227.
CSXT	4636-WATSEKA	IL	2278.
CSXT	4642-DANVILLE	IL	2323.
CSXT	3863-TERRE HAUTE	IN	2380.
CSXT	3812-VINCENNES	IN	2433.
CSXT	3838-EVANSVILLE	IN	2483.
CSXT	3839-HENDERSON	KY	2496.
CSXT	7061-HOPKINSVILLE	KY	2584.
CSXT	7201-MADISON	TN	2644.
CSXT	7202-NASHVILLE	TN	2654.
CSXT	7187-TULLAHOMA	TN	2733.
CSXT	7235-CHATTANOOGA	TN	2813.
CSXT	7888-DALTON	GA	2851.
CSXT	7889-CARTERSVILLE	GA	2902.
CSXT	7907-MARIETTA	GA	2935.
CSXT	7914-ATLANTA	GA	2944.
CSXT	7961-AUGUSTA	GA	3119.
CSXT	7732-ROBBINS	SC	3148.
CSXT	7717-DUNBARTON / WELL	SC	3157.
- - -	- - -	- - -	- - -
USG	7717-DUNBARTON / WELL	SC	3157.
USG	15359-SRP	SC	3165.

ROUTE FROM: BN 14100-TACOMA, WA
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	14179-PORTLAND	OR	150.
- - -	- - -	- - -	- - -
UP	14179-PORTLAND	OR	150.
UP	14197-OREGON TRUNK JCT	OR	245.
UP	14223-HINKLE	OR	337.
UP	14220-PENDLETON	OR	368.
UP	13412-NAMPA	ID	636.
UP	13370-POCATELLO	ID	878.
UP	13336-SCOVILLE	ID	934.

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	13964-KENNEWICK	WA	355.
BN	13890-PASCO	WA	357.
WCRC	13890-PASCO	WA	357.
WCRC	13964-KENNEWICK	WA	358.
WCRC	13941-RICHLAND	WA	366.
USG	13941-RICHLAND	WA	366.
USG	16212-HANFORD S 300	WA	374.

ROUTE FROM: BN 14100-TACOMA, WA
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14011-SEATTLE	WA	42.
BN	14008-EVERETT	WA	74.
BN	13828-SPOKANE	WA	383.
BN	13300-SANDPOINT	ID	447.
BN	13089-SHELBY	MT	784.
BN	13168-HAVRE	MT	885.
BN	15740-WILLISTON	ND	1204.
BN	10936-MINOT	ND	1316.
BN	10935-SURREY	ND	1322.
BN	11134-CASSELTON	ND	1537.
BN	11132-FARGO	ND	1557.
BN	11131-MOORHEAD	MN	1560.
BN	9663-STAPLES	MN	1674.
BN	9671-SAUK RAPIDS	MN	1739.
BN	9826-COON CREEK	MN	1789.
BN	9798-NORTHTOWN	MN	1797.
BN	9830-ST PAUL	MN	1809.
BN	5736-LA CROSSE	WI	1930.
BN	4327-EAST DUBUQUE	IL	2041.
BN	4317-SAVANNA	IL	2081.
BN	4190-AURORA	IL	2172.
BN	4170-LA GRANGE	IL	2197.
IHB	4170-LA GRANGE	IL	2197.
IHB	4172-ARGO	IL	2201.
IHB	4163-BLUE ISLAND	IL	2213.
IHB	4228-BURNHAM / CALUME	IL	2221.
NS	4228-BURNHAM / CALUME	IL	2221.
NS	4076-HAMMOND	IN	2223.
NS	4064-HOBART	IN	2240.
NS	4020-ARGOS	IN	2303.
NS	3548-FORT WAYNE	IN	2362.
NS	3650-MUNCIE	IN	2426.
NS	3688-RICHMOND	IN	2471.
NS	3251-HAMILTON	OH	2525.
NS	3234-IVORYDALE	OH	2542.
NS	3228-CINCINNATI	OH	2549.
NS	6850-LEXINGTON	KY	2623.
NS	6979-DANVILLE	KY	2660.
NS	7260-HARRIMAN	TN	2822.
NS	15316-K-25	TN	2837.

ROUTE FROM: BN 14100-TACOMA, WA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	14100-TACOMA	WA	0.
BN	14180-VANCOUVER	WA	142.
BN	14179-PORTLAND	OR	150.
UP	14179-PORTLAND	OR	150.
UP	14197-OREGON TRUNK JCT	OR	245.
UP	14223-HINKLE	OR	337.
UP	14220-PENDLETON	OR	368.
UP	13412-NAMPA	ID	636.
UP	13370-POCATELLO	ID	878.
UP	13369-MC CAMMON	ID	901.
UP	13568-OGDEN	UT	1015.
UP	13595-SALT LAKE CITY	UT	1050.
UP	13630-LYNNDYL	UT	1162.
UP	14766-VALLEY	NV	1479.
USG	14766-VALLEY	NV	1479.
USG	16333-YUCCA MOUNTAIN	NV	1578.

Figure E1-19 Representative Truck Routes from Wilmington, NC to Department of Energy Management Sites

From: WILMINGTON, NC
To : SRP, SC

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: WILMINGTON, NC
To : ID NATL ENG LAB, ID

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: WILMINGTON, NC
To : HANFORD, WA

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		DENVER	NE	I270	I70	CO
I270		COMMERCE CITY	NW	I270	I76	CO
I76		COMMERCE CITY	W	I25	I76	CO
I25		CHEYENNE	SW	I25	I80	WY
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: WILMINGTON, NC
To : K-25, TN

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: WILMINGTON, NC
To : MERCURY, NV

Routing through:

		WILMINGTON				NC
LOCAL		WILMINGTON	N	U117	LOCL	NC
U117		WILMINGTON	NW	U117	U421	NC
U421		WILMINGTON	W	U17	U421	NC
U17	U74	LELAND	SE	U17	U74	NC
U74	U76	CHADBOURN	NE	U74	U76	NC
U74		LUMBERTON	SW	I95	U74	NC
I95		FLORENCE	W	I20	I95	SC
I20		COLUMBIA	NW	I20	I26	SC
I26		ASHEVILLE	SW	I26	I40	NC
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		NASHVILLE	E	I24	I40	TN
I24	I40	NASHVILLE	SE	I24	I40	TN
I24	I65	INGLEWOOD	W	I24	I65	TN
I24		PULLEYS MILL	W	I24	I57	IL
I57		MT VERNON	SW	I57	I64	IL
I57	I64	MT VERNON	NW	I57	I64	IL
I64		WASHINGTON PK	SE	I255	I64	IL
I255		EDWARDSVILLE	SW	I255	I270	IL
I270		ST LOUIS	NW	I270	I70	MO
I70		KANSAS CITY	SE	I435	I70	MO
I435		KANSAS CITY	W	I435	I70	KS
I70 $	TKST$	TOPEKA	E	I470	I70	KS
I470$	TKST$	TOPEKA	S	I335	I470	KS
I470		TOPEKA	W	I470	I70	KS
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

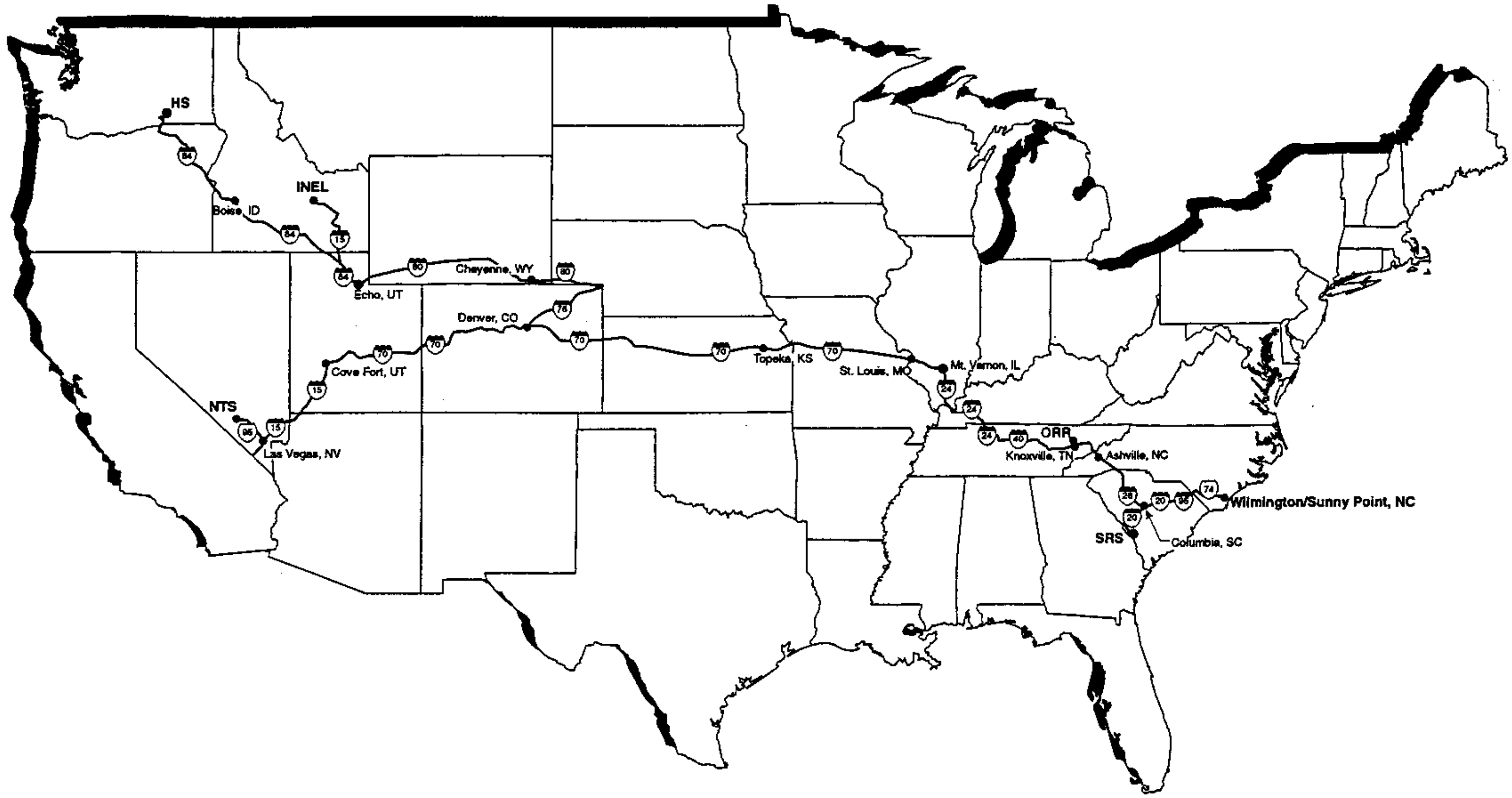


Figure E1-20 Representative Rail Routes from Wilmington, NC to Department of Energy Management Sites

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7671-DILLON	SC	106.
CSXT	7675-FLORENCE	SC	135.
CSXT	7690-CHARLESTON	SC	233.
CSXT	7739-FAIRFAX	SC	327.
CSXT	7732-ROBBINS	SC	356.
CSXT	7717-DUNBARTON / WELL	SC	365.
USG	7717-DUNBARTON / WELL	SC	365.
USG	15359-SRP	SC	373.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7407-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13369-MC CAMMON	ID	2570.
UP	13370-POCATELLO	ID	2593.
UP	13336-SCOVILLE	ID	2649.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7407-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13369-MC CAMMON	ID	2570.
UP	13370-POCATELLO	ID	2593.
UP	13412-NAMPA	ID	2835.
UP	14220-PENDLETON	OR	3103.
UP	14223-HINKLE	OR	3134.
UP	13894-WALLULA	WA	3163.
UP	13964-KENNEWICK	WA	3178.
UP	13941-RICHLAND	WA	3187.
USG	13941-RICHLAND	WA	3187.
USG	16212-HANFORD S 300	WA	3195.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7472-WADESBORO	NC	142.
WSS	7472-WADESBORO	NC	142.
WSS	7462-LEXINGTON	NC	210.
NS	7462-LEXINGTON	NC	210.
NS	7478-SALISBURY	NC	227.
NS	7394-HICKORY	NC	284.
NS	7387-MARION	NC	326.
NS	7343-ASHEVILLE	NC	366.
NS	7318-MORRISTOWN	TN	446.
NS	7286-KNOXVILLE	TN	487.
NS	7288-DOSSETT	TN	512.
NS	15316-K-25	TN	533.

ROUTE FROM: CSXT 7625-WILMINGTON, NC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CSXT	7625-WILMINGTON	NC	0.
CSXT	7620-PEMBROKE	NC	86.
CSXT	7470-HAMLET	NC	117.
CSXT	7407-MONROE	NC	170.
CSXT	7834-CLINTON	SC	262.
CSXT	7838-GREENWOOD	SC	290.
CSXT	7956-ATHENS	GA	371.
CSXT	7914-ATLANTA	GA	462.
CSXT	7907-MARIETTA	GA	472.
CSXT	7889-CARTERSVILLE	GA	504.
CSXT	7888-DALTON	GA	555.
CSXT	7235-CHATTANOOGA	TN	593.
CSXT	7187-TULLAHOMA	TN	674.
CSXT	7202-NASHVILLE	TN	753.
CSXT	7201-MADISON	TN	763.
CSXT	7061-HOPKINSVILLE	KY	823.
CSXT	3839-HENDERSON	KY	910.
CSXT	3838-EVANSVILLE	IN	923.
CSXT	3812-VINCENNES	IN	973.
CSXT	4952-SALEM	IL	1052.
CSXT	10859-EAST ST LOUIS	IL	1117.
TRRA	10859-EAST ST LOUIS	IL	1117.
TRRA	10858-ST LOUIS	MO	1123.
UP	10858-ST LOUIS	MO	1123.
UP	10656-JEFFERSON CITY	MO	1245.
UP	10616-KANSAS CITY	MO	1421.
UP	10617-KANSAS CITY	KS	1424.
UP	11823-LAWRENCE	KS	1463.
UP	11697-TOPEKA	KS	1493.
UP	11696-MENOKEN	KS	1498.
UP	11681-MARYSVILLE	KS	1573.
UP	11405-HASTINGS	NE	1683.
UP	11410-GIBBON	NE	1709.
UP	11352-NORTH PLATTE	NE	1787.
UP	11358-O FALLONS	NE	1836.
UP	13703-JULESBURG	CO	1904.
UP	13465-CHEYENNE	WY	2050.
UP	13462-LARAMIE	WY	2102.
UP	13494-GRANGER	WY	2378.
UP	13568-OGDEN	UT	2517.
UP	13595-SALT LAKE CITY	UT	2552.
UP	13630-LYNNDYL	UT	2664.
UP	14766-VALLEY	NV	2981.
USG	14766-VALLEY	NV	2981.
USG	16333-YUCCA MOUNTAIN	NV	3080.

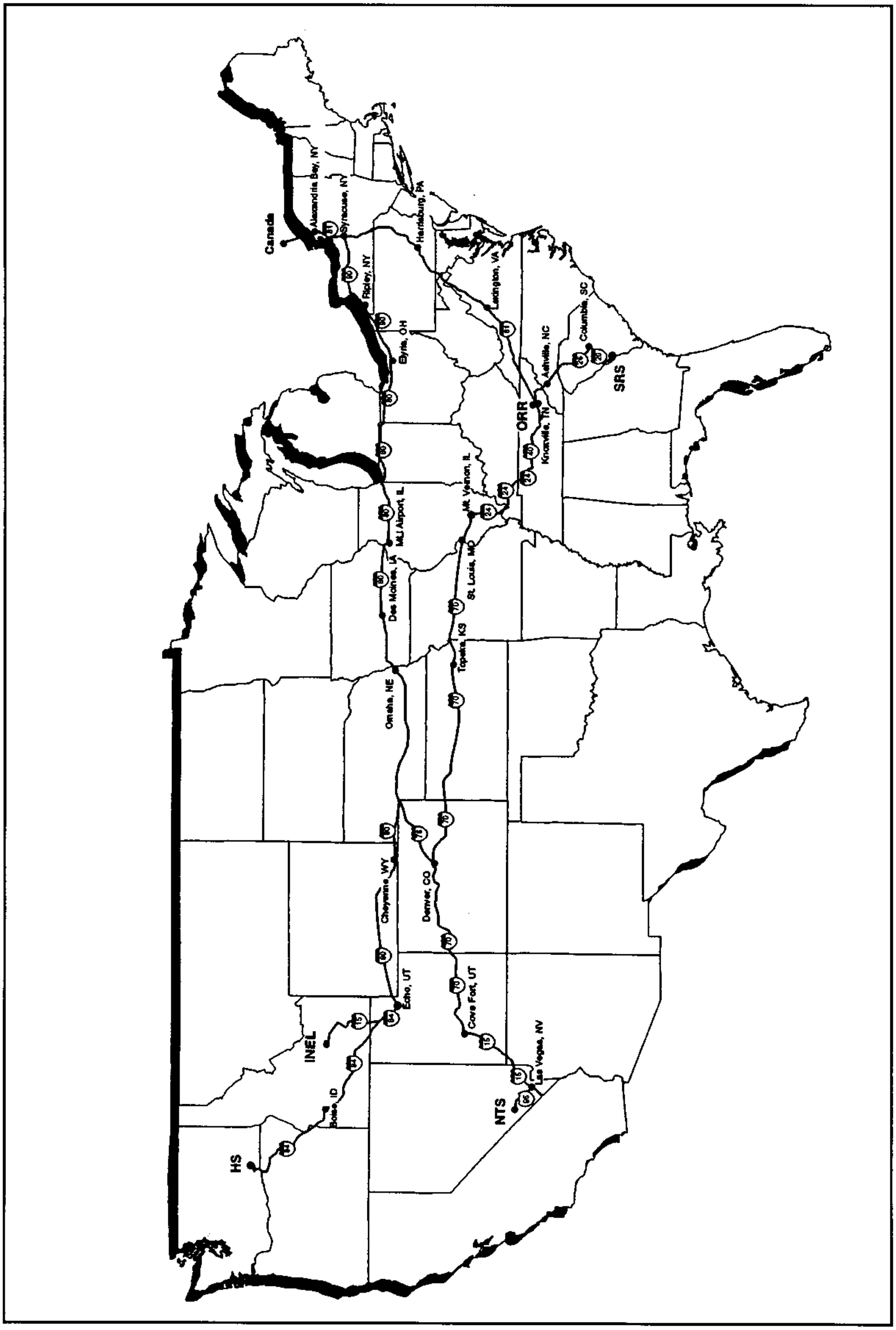


Figure E1-21 Representative Truck Routes from Eastern Canada to Department of Energy Management Sites

From: ALEXANDRIA BAY SW I81 S12, NY
To : SRP, SC

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		CICERO	S	I481	I81	NY
I481		SYRACUSE	S	I481	I81	NY
I81		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77		COLUMBIA	NE	I20	I77	SC
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRP				SC

From: ALEXANDRIA BAY SW I81 S12, NY
To : ID NATL ENG LAB, ID

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	BUFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGHBY HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PORTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I80		DES MOINES	N	I235	I35	IA
I35	I80	DES MOINES	W	I235	I35	IA
I80		MINDEN	NW	I680	I80	IA
I680		LOVELAND	SW	I29	I680	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: ALEXANDRIA BAY SW I81 S12, NY
To : HANFORD, WA

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	BUFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGHBY HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PORTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I80		DES MOINES	N	I235	I35	IA
I35	I80	DES MOINES	W	I235	I35	IA
I80		MINDEN	NW	I680	I80	IA
I680		LOVELAND	SW	I29	I680	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		ECHO		I80	I84	UT
I84		OGDEN	S	I15	I84	UT
I15	I84	TREMONTON	W	I15	I84	UT
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: ALEXANDRIA BAY SW I81 S12, NY
To : K-25, TN

Routing through:

		ALEXANDRIA BAY	SW	I81	S12	NY
I81		CICERO	S	I481	I81	NY
I481		SYRACUSE	S	I481	I81	NY
I81		STAUNTON	SE	I64	I81	VA
I64	I81	LEXINGTON	E	I64	I81	VA
I81		FT CHISWELL	E	I77	I81	VA
I77	I81	WYTHEVILLE	E	I77	I81	VA
I81		DANDRIDGE	NE	I40	I81	TN
I40		KNOXVILLE	NE	I40	I640	TN
I640		KNOXVILLE	NW	I640	I75	TN
I640	I75	KNOXVILLE	W	I40	I640	TN
I40	I75	OAK RIDGE	S	I40	I75	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: ALEXANDRIA BAY SW I81 S12, NY
To : YUCCA MOUNTAIN, NV

Routing through:

	ALEXANDRIA BAY	SW	I81	S12	NY
I81		SYRACUSE	N	I81	I90	NY
I90 $	TNYT$	BUFFALO	NE	I290	I90	NY
I90	TNYT	LACKAWANNA	E	I90	X55	NY
I90 $	TNYT$	RIPLEY	W	I90	X61	NY
I90		WILLOUGHBY HLS	W	I271	I90	OH
I271		BEDFORD	NE	I271	I480	OH
I480		N RIDGEVILLE	S	I80	X9A	OH
I80 $		ELYRIA	NW	I80	I90	OH
I80 $	I90 $	PORTAGE	W	I80	I90	IN
I80	I94	LANSING	W	I294	I80	IL
I294$	I80 $	HOMEWOOD	NW	I294	I80	IL
I80		GREEN ROCK	SE	I74	I80	IL
I280	I74	MLI AIRPORT		I280	I74	IL
I280		DAVENPORT	NW	I280	I80	IA
I80		DES MOINES	N	I235	I35	IA
I35	I80	DES MOINES	W	I235	I35	IA
I80		MINDEN	NW	I680	I80	IA
I680		LOVELAND	SW	I29	I680	IA
I29	I680	CRESCENT	W	I29	I680	IA
I680		OMAHA	SW	I680	I80	NE
I80		BIG SPRINGS	SW	I76	I80	NE
I76		COMMERCE CITY	W	I25	I76	CO
I25		DENVER	N	I25	I70	CO
I70		COVE FORT	W	I15	I70	UT
I15		LAS VEGAS				NV
U95		AMARGOSA VALLY		U95	S373	NV
LOCAL		YUCCA MOUNTAIN				NV

Figure E1-22 Representative Rail Routes from Eastern Canada to Department of Energy Management Sites

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	755-ROME	NY	145.
CR	756-UTICA	NY	159.
CR	706-SCHENECTADY	NY	237.
CR	700-SELKIRK	NY	259.
CR	1094-NEWBURGH	NY	334.
CR	1215-JERSEY CITY	NJ	400.
CR	1183-NEWARK	NJ	405.
CR	1230-ALDENE	NJ	411.
CR	1311-BOUND BROOK	NJ	430.
CR	1447-PHILADELPHIA	PA	492.
CR	2456-WILMINGTON	DE	529.
CR	2516-BALTIMORE	MD	593.
CR	2596-WASHINGTON	DC	627.
CR	2595-ALEXANDRIA	VA	637.
CSXT	2595-ALEXANDRIA	VA	637.
CSXT	6082-RICHMOND	VA	752.
CSXT	6087-COLONIAL HEIGHTS	VA	774.
CSXT	6064-PETERSBURG	VA	779.
CSXT	7563-WELDON	NC	839.
CSXT	7565-ROCKY MOUNT	NC	876.
CSXT	7566-WILSON	NC	890.
CSXT	7606-FAYETTEVILLE	NC	964.
CSXT	7620-PEMBROKE	NC	993.
CSXT	7671-DILLON	SC	1013.
CSXT	7675-FLORENCE	SC	1042.
CSXT	7690-CHARLESTON	SC	1140.
CSXT	7739-FAIRFAX	SC	1234.
CSXT	7732-ROBBINS	SC	1263.
CSXT	7717-DUNBARTON / WELL	SC	1272.
USG	7717-DUNBARTON / WELL	SC	1272.
USG	15359-SRP	SC	1280.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INDIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-DE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEDAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVADA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLAND	NE	1365.
UP	11410-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULESBURG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13369-MC CAMMON	ID	2252.
UP	13370-POCATELLO	ID	2275.
UP	13336-SCOVILLE	ID	2331.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INDIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-DE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEDAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVADA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLAND	NE	1365.
UP	11410-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULESBURG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13369-MC CAMMON	ID	2252.
UP	13370-POCATELLO	ID	2275.
UP	13412-NAMPA	ID	2517.
UP	14220-PENDLETON	OR	2786.
UP	14223-HINKLE	OR	2817.
UP	13894-WALLULA	WA	2846.
UP	13964-KENNEWICK	WA	2861.
UP	13941-RICHLAND	WA	2870.
USG	13941-RICHLAND	WA	2870.
USG	16212-HANFORD S 300	WA	2877.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2629-WELLINGTON	OH	466.
CR	3399-CRESTLINE	OH	504.
CR	3094-COLUMBUS (4TH ST	OH	566.
CR	3095-COLUMBUS (BROAD	OH	567.
CR	14993-COLUMBUS (BUCKEY	OH	569.
CR	3300-SPRINGFIELD	OH	611.
CR	3282-DAYTON	OH	632.
CR	3250-MIDDLETOWN	OH	652.
CR	3234-IVORYDALE	OH	677.
CR	3228-CINCINNATI	OH	684.
NS	3228-CINCINNATI	OH	684.
NS	6850-LEXINGTON	KY	758.
NS	6979-DANVILLE	KY	795.
NS	7260-HARRIMAN	TN	957.
NS	15316-K-25	TN	972.

ROUTE FROM: CR 738-GOUVERNEUR, NY
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
CR	738-GOUVERNEUR	NY	0.
CR	777-SYRACUSE	NY	100.
CR	780-SOLVAY	NY	103.
CR	817-ROCHESTER	NY	179.
CR	880-BUFFALO	NY	241.
CR	968-ERIE	PA	333.
CR	2649-ASHTABULA	OH	372.
CR	2728-CLEVELAND	OH	429.
CR	2633-ELYRIA	OH	456.
CR	3442-TOLEDO	OH	535.
CR	3526-GOSHEN	IN	657.
CR	3525-ELKHART	IN	667.
CR	4022-SOUTH BEND	IN	682.
CR	4067-PORTER	IN	727.
CR	4070-GARY	IN	742.
CR	4073-CLARKE	IN	746.
CR	4074-INDIANA HARBOR	IN	749.
CR	4232-SOUTH CHICAGO	IL	757.
CR	4217-CHICAGO	IL	770.
CNW	4217-CHICAGO	IL	770.
CNW	4234-PROVISO	IL	784.
CNW	4311-DE KALB	IL	826.
CNW	4324-NELSON	IL	871.
CNW	10304-CLINTON	IA	903.
CNW	10289-CEDAR RAPIDS	IA	984.
CNW	10265-MARSHALLTOWN	IA	1051.
CNW	10246-NEVADA	IA	1078.
CNW	10271-AMES	IA	1089.
CNW	10176-MISSOURI VALLEY	IA	1222.
CNW	10198-CALIFORNIA JCT	IA	1228.
CNW	11340-FREMONT	NE	1256.
UP	11340-FREMONT	NE	1256.
UP	11406-GRAND ISLAND	NE	1365.
UP	11410-GIBBON	NE	1391.
UP	11352-NORTH PLATTE	NE	1469.
UP	11358-O FALLONS	NE	1518.
UP	13703-JULESBURG	CO	1586.
UP	13465-CHEYENNE	WY	1732.
UP	13462-LARAMIE	WY	1784.
UP	13494-GRANGER	WY	2060.
UP	13568-OGDEN	UT	2199.
UP	13595-SALT LAKE CITY	UT	2235.
UP	13630-LYNNDYL	UT	2347.
UP	14766-VALLEY	NV	2664.
USG	14766-VALLEY	NV	2664.
USG	16333-YUCCA MOUNTAIN	NV	2763.

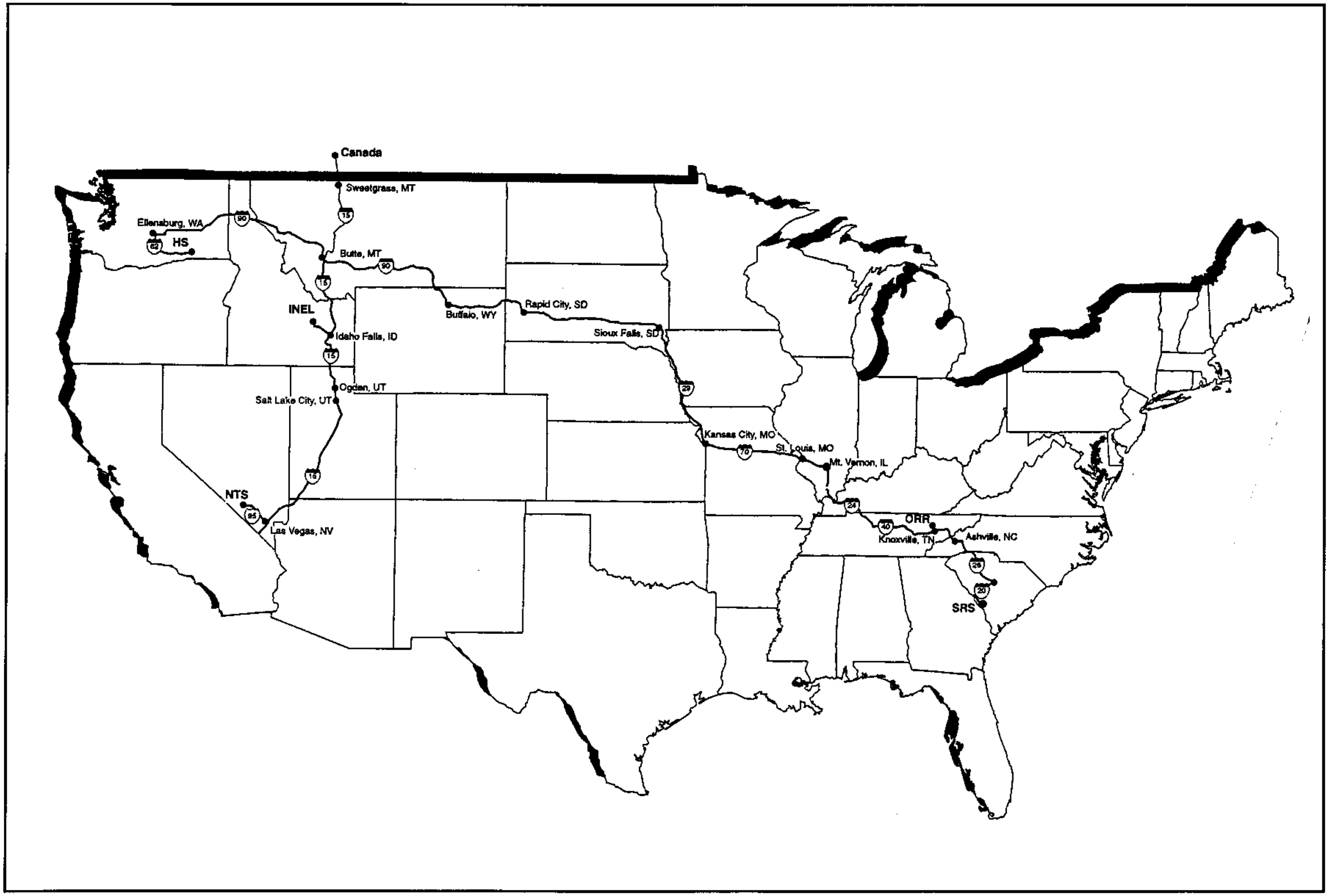


Figure E1-23 Representative Truck Routes from Western Canada to Department of Energy Management Sites

From: SWEETGRASS I15 X394 MT
To : SRL, SC

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	N	I24	I265	TN
I265		NASHVILLE	W	I265	I40	TN
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: SWEETGRASS I15 X394 MT
To : ID NATL ENG LAB, ID

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I15		BLACKFOOT	NW	I15	X92	ID
U26		ATOMIC CITY	NW	U20	U26	ID
U20	U26	ID NATL ENG LAB				ID

From: SWEETGRASS I15 X394 MT
To : HANFORD, WA

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I90		ELLENSBURG	SE	I82	I90	WA
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: SWEETGRASS I15 X394 MT
To : K-25, TN

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I90		SIOUX FALLS	NW	I29	I90	SD
I29		LOVELAND	SW	I29	I680	IA
I680		MINDEN	NW	I680	I80	IA
I80		COUNCIL BLUFFS	SE	I29	I80	IA
I29		KANSAS CITY	NW	I29	I435	MO
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	N	I24	I265	TN
I265		NASHVILLE	W	I265	I40	TN
I40		NASHVILLE	W	I40	I440	TN
I440		NASHVILLE	SE	I24	I440	TN
I24		NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: SWEETGRASS I15 X394 MT
To : MERCURY, NV

Routing through:

		SWEETGRASS		I15	X394	MT
I15		BUTTE	E	I15	I90	MT
I15	I90	BUTTE	W	I15	I90	MT
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		N SALT LAKE		I15	I215	UT
I215		MIDVALE		I15	I215	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

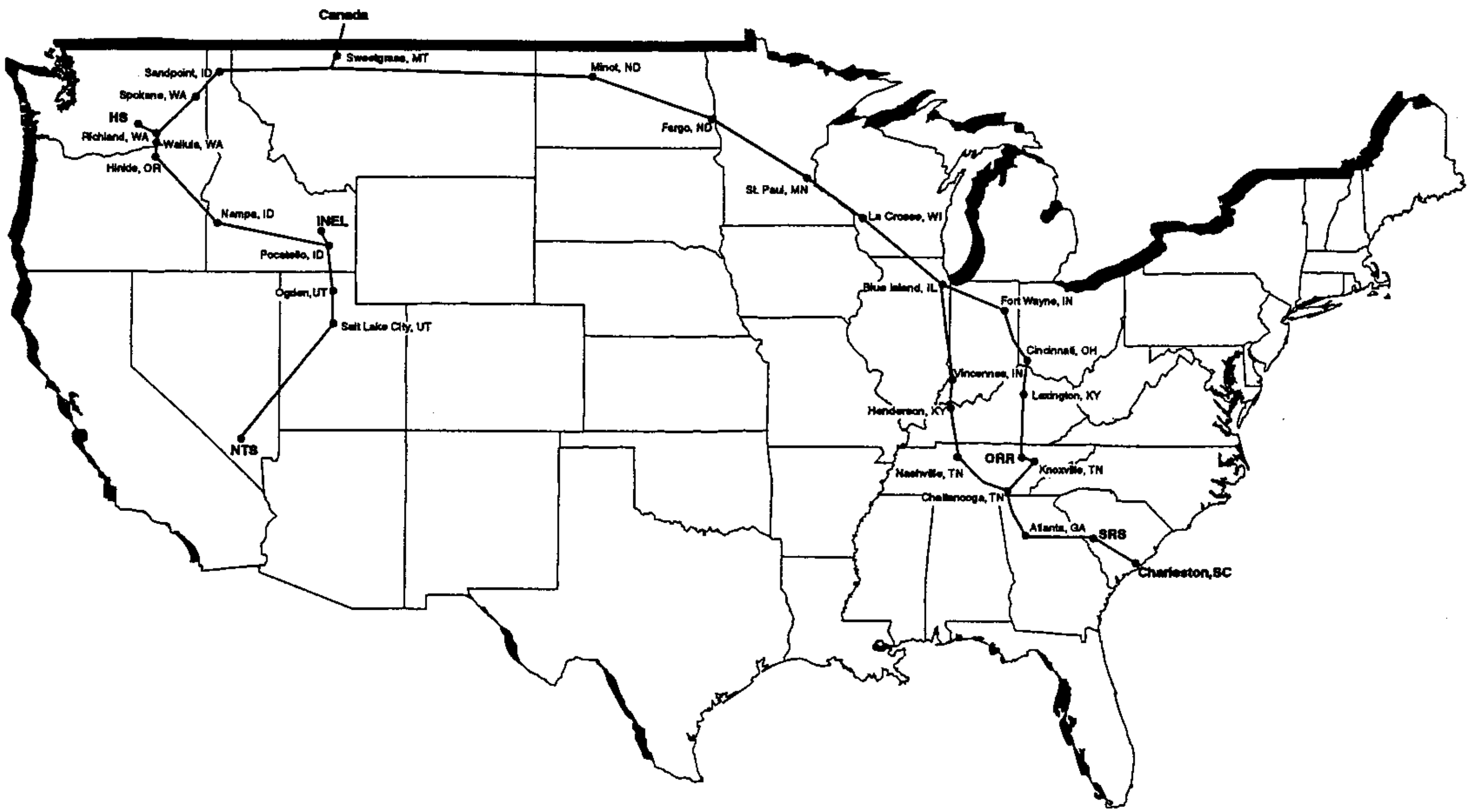


Figure E1-24 Representative Rail Routes from Western Canada to Department of Energy Management Sites

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13089-SHELBY	MT	43.
BN	13168-HAVRE	MT	144.
BN	15740-WILLISTON	ND	463.
BN	10936-MINOT	ND	575.
BN	10935-SURREY	ND	581.
BN	11134-CASSELTON	ND	796.
BN	11132-FARGO	ND	816.
BN	11131-MOORHEAD	MN	819.
BN	9663-STAPLES	MN	933.
BN	9671-SAUK RAPIDS	MN	998.
BN	9826-COON CREEK	MN	1048.
BN	9798-NORTHTOWN	MN	1056.
BN	9830-ST PAUL	MN	1069.
BN	5736-LA CROSSE	WI	1190.
BN	4327-EAST DUBUQUE	IL	1301.
BN	4317-SAVANNA	IL	1341.
BN	4190-AURORA	IL	1432.
BN	4170-LA GRANGE	IL	1457.
IHB	4170-LA GRANGE	IL	1457.
IHB	4172-ARGO	IL	1461.
IHB	4163-BLUE ISLAND	IL	1473.
IHB	4223-DOLTON / RIVERDAIL		1477.
CSXT	4223-DOLTON / RIVERDAIL		1477.
CSXT	4206-CHICAGO HEIGHTS	IL	1487.
CSXT	4636-WATSEKA	IL	1538.
CSXT	4642-DANVILLE	IL	1583.
CSXT	3863-TERRE HAUTE	IN	1640.
CSXT	3812-VINCENNES	IN	1693.
CSXT	3838-EVANSVILLE	IN	1743.
CSXT	3839-HENDERSON	KY	1756.
CSXT	7061-HOPKINSVILLE	KY	1843.
CSXT	7201-MADISON	TN	1903.
CSXT	7202-NASHVILLE	TN	1913.
CSXT	7187-TULLAHOMA	TN	1992.
CSXT	7235-CHATTANOOGA	TN	2073.
CSXT	7888-DALTON	GA	2111.
CSXT	7889-CARTERSVILLE	GA	2162.
CSXT	7907-MARIETTA	GA	2194.
CSXT	7914-ATLANTA	GA	2204.
CSXT	7961-AUGUSTA	GA	2379.
CSXT	7732-ROBBINS	SC	2408.
CSXT	7717-DUNBARTON / WELL	SC	2417.
USG	7717-DUNBARTON / WELL	SC	2417.
USG	15359-SRP	SC	2425.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANDPOINT	ID	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
BN	13894-WALLULA	WA	605.
UP	13894-WALLULA	WA	605.
UP	14223-HINKLE	OR	634.
UP	14220-PENDLETON	OR	665.
UP	13412-NAMPA	ID	933.
UP	13370-POCATELLO	ID	1175.
UP	13336-SCOVILLE	ID	1231.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANDPOINT	ID	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
WCRC	13890-PASCO	WA	589.
WCRC	13964-KENNEWICK	WA	590.
WCRC	13941-RICHLAND	WA	598.
USG	13941-RICHLAND	WA	598.
USG	16212-HANFORD S 300	WA	606.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: NS 15316-K-25, TN

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13089-SHELBY	MT	43.
BN	13168-HAVRE	MT	144.
BN	15740-WILLISTON	ND	463.
BN	10936-MINOT	ND	575.
BN	10935-SURREY	ND	581.
BN	11134-CASSELTON	ND	796.
BN	11132-FARGO	ND	816.
BN	11131-MOORHEAD	MN	819.
BN	9663-STAPLES	MN	933.
BN	9671-SAUK RAPIDS	MN	998.
BN	9826-COON CREEK	MN	1048.
BN	9798-NORTHTOWN	MN	1056.
BN	9830-ST PAUL	MN	1069.
BN	5736-LA CROSSE	WI	1190.
BN	4327-EAST DUBUQUE	IL	1301.
BN	4317-SAVANNA	IL	1341.
BN	4190-AURORA	IL	1432.
BN	4170-LA GRANGE	IL	1457.
IHB	4170-LA GRANGE	IL	1457.
IHB	4172-ARGO	IL	1461.
IHB	4163-BLUE ISLAND	IL	1473.
IHB	4228-BURNHAM / CALUME	IL	1481.
NS	4228-BURNHAM / CALUME	IL	1481.
NS	4076-HAMMOND	IN	1483.
NS	4064-HOBART	IN	1499.
NS	4020-ARGOS	IN	1562.
NS	3548-FORT WAYNE	IN	1621.
NS	3650-MUNCIE	IN	1685.
NS	3688-RICHMOND	IN	1730.
NS	3251-HAMILTON	OH	1785.
NS	3234-IVORYDALE	OH	1802.
NS	3228-CINCINNATI	OH	1809.
NS	6850-LEXINGTON	KY	1883.
NS	6979-DANVILLE	KY	1920.
NS	7260-HARRIMAN	TN	2082.
NS	15316-K-25	TN	2097.

ROUTE FROM: BN 13066-SWEET GRASS, MT
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
BN	13066-SWEET GRASS	MT	0.
BN	13300-SANDPOINT	ID	374.
BN	13828-SPOKANE	WA	437.
BN	13890-PASCO	WA	589.
BN	13894-WALLULA	WA	605.
UP	13894-WALLULA	WA	605.
UP	14223-HINKLE	OR	634.
UP	14220-PENDLETON	OR	665.
UP	13412-NAMPA	ID	933.
UP	13370-POCATELLO	ID	1175.
UP	13369-MC CAMMON	ID	1198.
UP	13568-OGDEN	UT	1312.
UP	13595-SALT LAKE CITY	UT	1347.
UP	13630-LYNNDYL	UT	1459.
UP	14766-VALLEY	NV	1776.
USG	14766-VALLEY	NV	1776.
USG	16333-YUCCA MOUNTAIN	NV	1875.

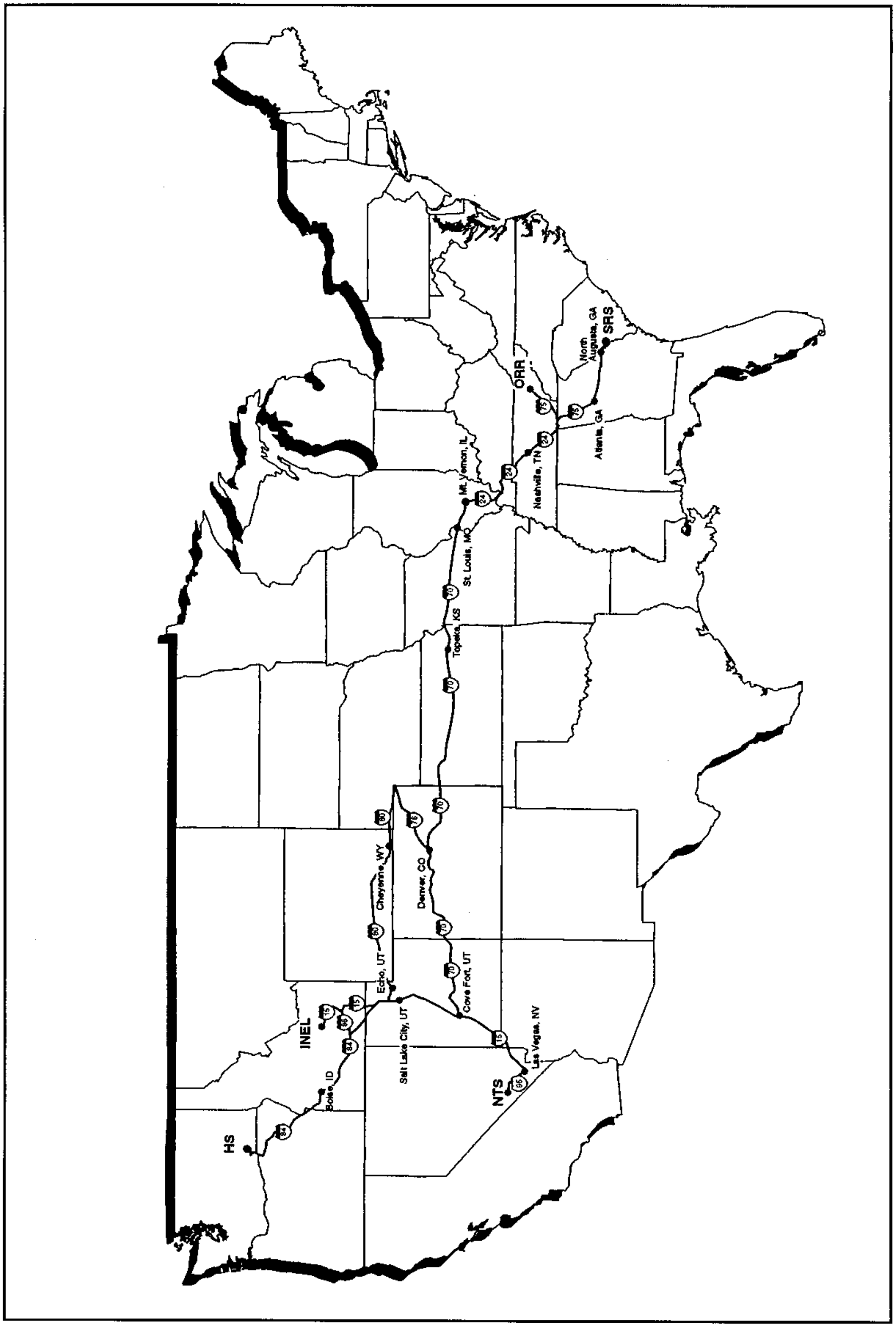


Figure E1-25 Representative Truck Routes Between Department of Energy Management Sites

From: ID NATL ENG LAB, ID
To : HANFORD, WA

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		CHUBBUCK	E	I15	I86	ID
I86		RAFT RIVER	W	I84	I86	ID
I84		HERMISTON	SW	I82	I84	OR
I82		WEST RICHLAND	S	I182	I82	WA
I182		RICHLAND	SE	I182	S240	WA
S240		RICHLAND	N	S240	LR4S	WA
LR4S		HANFORD				WA

From: ID NATL ENG LAB, ID
To : MERCURY, NV

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: ID NATL ENG LAB, ID
To : SRL, SC

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: ID NATL ENG LAB, ID
To : K-25, TN

Routing through:

		ID NATL ENG LAB				ID
U20	U26	ATOMIC CITY	NW	U20	U26	ID
U26		BLACKFOOT	NW	I15	X92	ID
I15		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: HANFORD, WA
To : MERCURY, NV

Routing through:

		HANFORD				WA
LR4S		RICHLAND	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I15		SALT LAKE CITY	W	I15	I80	UT
I15	I80	SALT LAKE CITY	S	I15	I80	UT
I15		LAS VEGAS				NV
U95		MERCURY	S	U95	LOCL	NV
LOCAL		MERCURY				NV

From: HANFORD, WA
To : SRL, SC

Routing through:

		HANFORD				WA
LR4S		RICHLAND	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: HANFORD, WA
To : K-25, TN

Routing through:

		HANFORD				WA
LR4S		RICHLAND	N	S240	LR4S	WA
S240		RICHLAND	SE	I182	S240	WA
I182		WEST RICHLAND	S	I182	I82	WA
I82		HERMISTON	SW	I82	I84	OR
I84		TREMONTON	W	I15	I84	UT
I15	I84	OGDEN	S	I15	I84	UT
I84		ECHO		I80	I84	UT
I80		CHEYENNE	SW	I25	I80	WY
I25		COMMERCE CITY	W	I25	I76	CO
I76		COMMERCE CITY	NW	I270	I76	CO
I270		DENVER	NE	I270	I70	CO
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

From: K-25, TN
To : SRL, SC

Routing through:

		K-25				TN
S58		KINGSTON	E	I40	S58	TN
I40		OAK RIDGE	S	I40	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: MERCURY, NV
To : SRL, SC

Routing through:

		MERCURY				NV
LOCAL		MERCURY	S	U95	LOCL	NV
U95		LAS VEGAS				NV
I15		COVE FORT	W	I15	I70	UT
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I24		EAST RIDGE	NE	I24	I75	TN
I75		ATLANTA	NW	I285	I75	GA
I285		ATLANTA	E	I20	I285	GA
I20		NORTH AUGUSTA	NW	I20	S230	SC
S230		NORTH AUGUSTA				SC
S125		CLEARWATER	W	U1	U278	SC
U278		BEECH ISLAND		U278	S125	SC
S125		JACKSON	SE	S125	LSRP	SC
LSRP		SRL				SC

From: MERCURY, NV
To : K-25, TN

Routing through:

		MERCURY				NV
LOCAL		MERCURY	S	U95	LOCL	NV
U95		LAS VEGAS				NV
I15		COVE FORT	W	I15	I70	UT
I70		TOPEKA	W	I470	I70	KS
I470		TOPEKA	S	I335	I470	KS
I470$	TKST$	TOPEKA	E	I470	I70	KS
I70 $	TKST$	KANSAS CITY	W	I435	I70	KS
I435		KANSAS CITY	SE	I435	I70	MO
I70		ST LOUIS	NW	I270	I70	MO
I270		EDWARDSVILLE	SW	I255	I270	IL
I255		WASHINGTON PK	SE	I255	I64	IL
I64		MT VERNON	NW	I57	I64	IL
I57	I64	MT VERNON	SW	I57	I64	IL
I57		PULLEYS MILL	W	I24	I57	IL
I24		INGLEWOOD	W	I24	I65	TN
I24	I65	NASHVILLE	SE	I24	I40	TN
I24	I40	NASHVILLE	E	I24	I40	TN
I40		KINGSTON	E	I40	S58	TN
S58		K-25				TN

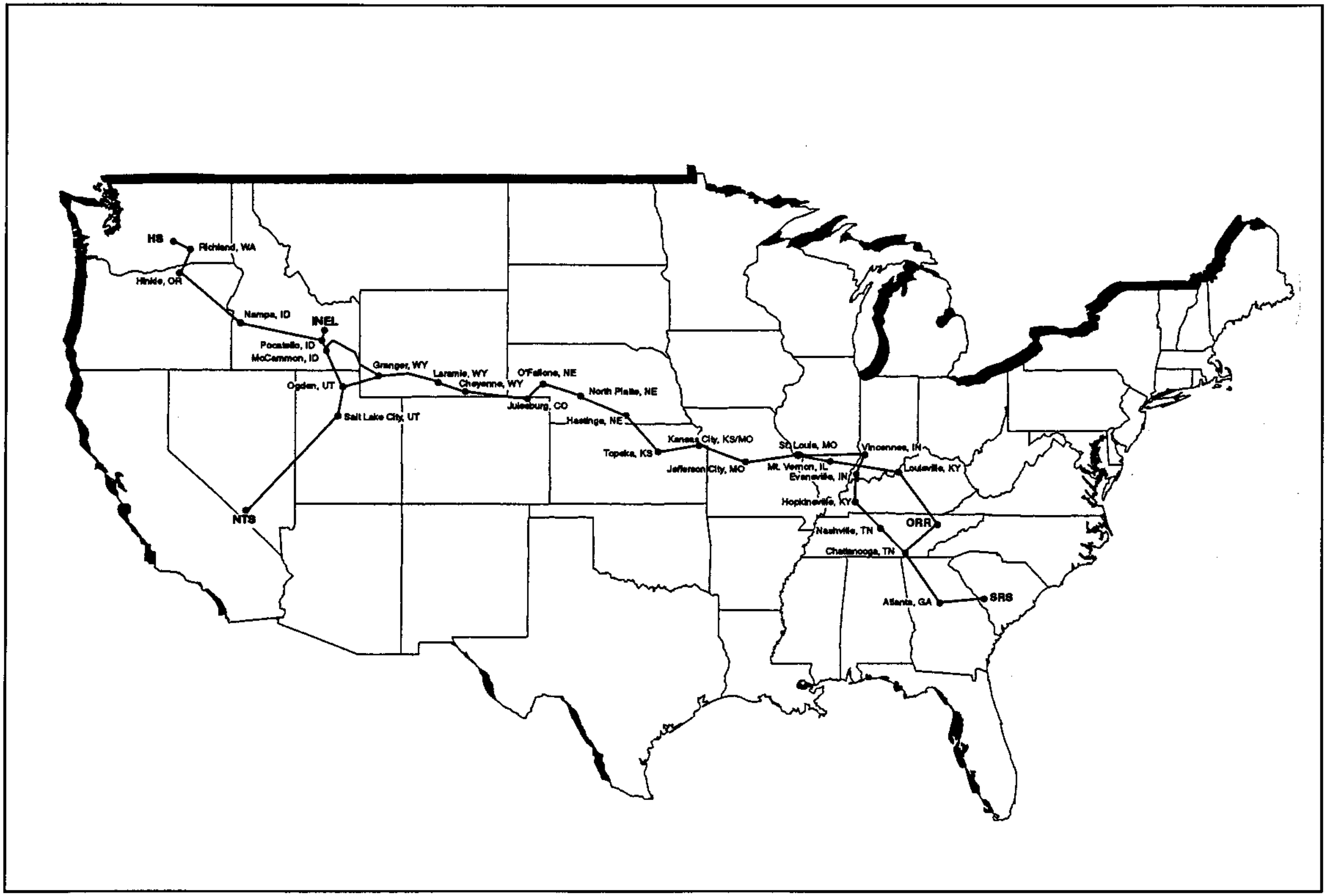


Figure E1-26 Representative Rail Routes Between Department of Energy Management Sites

ROUTE FROM: USG 15359-SRP, SC
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13369-MC CAMMON	ID	2329.
UP	13370-POCATELLO	ID	2352.
UP	13412-NAMPA	ID	2594.
UP	14220-PENDLETON	OR	2862.
UP	14223-HINKLE	OR	2893.
UP	13894-WALLULA	WA	2922.
UP	13964-KENNEWICK	WA	2937.
UP	13941-RICHLAND	WA	2946.
USG	13941-RICHLAND	WA	2946.
USG	16212-HANFORD S 300	WA	2954.

ROUTE FROM: USG 15359-SRP, SC
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13369-MC CAMMON	ID	2329.
UP	13370-POCATELLO	ID	2352.
UP	13336-SCOVILLE	ID	2408.

ROUTE FROM: USG 15359-SRP, SC
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
USG	15359-SRP	SC	0.
USG	7717-DUNBARTON / WELL	SC	8.
CSXT	7717-DUNBARTON / WELL	SC	8.
CSXT	7732-ROBBINS	SC	17.
CSXT	7961-AUGUSTA	GA	46.
CSXT	7914-ATLANTA	GA	221.
CSXT	7907-MARIETTA	GA	231.
CSXT	7889-CARTERSVILLE	GA	263.
CSXT	7888-DALTON	GA	314.
CSXT	7235-CHATTANOOGA	TN	352.
CSXT	7187-TULLAHOMA	TN	433.
CSXT	7202-NASHVILLE	TN	512.
CSXT	7201-MADISON	TN	522.
CSXT	7061-HOPKINSVILLE	KY	582.
CSXT	3839-HENDERSON	KY	669.
CSXT	3838-EVANSVILLE	IN	682.
CSXT	3812-VINCENNES	IN	732.
CSXT	4952-SALEM	IL	811.
CSXT	10859-EAST ST LOUIS	IL	876.
<TR>	10859-EAST ST LOUIS	IL	876.
<TR>	10858-ST LOUIS	MO	882.
UP	10858-ST LOUIS	MO	882.
UP	10656-JEFFERSON CITY	MO	1004.
UP	10616-KANSAS CITY	MO	1180.
UP	10617-KANSAS CITY	KS	1183.
UP	11823-LAWRENCE	KS	1222.
UP	11697-TOPEKA	KS	1252.
UP	11696-MENOKEN	KS	1257.
UP	11681-MARYSVILLE	KS	1332.
UP	11405-HASTINGS	NE	1442.
UP	11410-GIBBON	NE	1468.
UP	11352-NORTH PLATTE	NE	1546.
UP	11358-O FALLONS	NE	1595.
UP	13703-JULESBURG	CO	1663.
UP	13465-CHEYENNE	WY	1809.
UP	13462-LARAMIE	WY	1861.
UP	13494-GRANGER	WY	2137.
UP	13568-OGDEN	UT	2276.
UP	13595-SALT LAKE CITY	UT	2311.
UP	13630-LYNNDYL	UT	2423.
UP	14766-VALLEY	NV	2740.
USG	14766-VALLEY	NV	2740.
USG	16333-YUCCA MOUNTAIN	NV	2839.

ROUTE FROM: UP 13336-SCOVILLE, ID
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
UP	13336-SCOVILLE	ID	0.
UP	13370-POCATELLO	ID	56.
UP	13412-NAMPA	ID	298.
UP	14220-PENDLETON	OR	567.
UP	14223-HINKLE	OR	598.
UP	13894-WALLULA	WA	627.
UP	13964-KENNEWICK	WA	642.
UP	13941-RICHLAND	WA	650.
USG	13941-RICHLAND	WA	650.
USG	16212-HANFORD S 300	WA	658.

ROUTE FROM: UP 13336-SCOVILLE, ID
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
UP	13336-SCOVILLE	ID	0.
UP	13370-POCATELLO	ID	56.
UP	13369-MC CAMMON	ID	79.
UP	13568-OGDEN	UT	193.
UP	13595-SALT LAKE CITY	UT	228.
UP	13630-LYNNDYL	UT	340.
UP	14766-VALLEY	NV	657.
USG	14766-VALLEY	NV	657.
USG	16333-YUCCA MOUNTAIN	NV	756.

ROUTE FROM: USG 16212-HANFORD S 300, WA
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
USG	16212-HANFORD S 300	WA	0.
USG	13941-RICHLAND	WA	8.
UP	13941-RICHLAND	WA	8.
UP	13964-KENNEWICK	WA	16.
UP	13894-WALLULA	WA	31.
UP	14223-HINKLE	OR	60.
UP	14220-PENDLETON	OR	91.
UP	13412-NAMPA	ID	360.
UP	13370-POCATELLO	ID	602.
UP	13369-MC CAMMON	ID	625.
UP	13568-OGDEN	UT	738.
UP	13595-SALT LAKE CITY	UT	774.
UP	13630-LYNNDYL	UT	886.
UP	14766-VALLEY	NV	1203.
USG	14766-VALLEY	NV	1203.
USG	16333-YUCCA MOUNTAIN	NV	1302.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 16212-HANFORD S 300, WA

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13369-MC CAMMON	ID	1976.
UP	13370-POCATELLO	ID	1999.
UP	13412-NAMPA	ID	2241.
UP	14220-PENDLETON	OR	2510.
UP	14223-HINKLE	OR	2541.
UP	13894-WALLULA	WA	2570.
UP	13964-KENNEWICK	WA	2585.
UP	13941-RICHLAND	WA	2594.
USG	13941-RICHLAND	WA	2594.
USG	16212-HANFORD S 300	WA	2601.

ROUTE FROM: NS 15316-K-25, TN
TO: UP 13336-SCOVILLE, ID

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13369-MC CAMMON	ID	1976.
UP	13370-POCATELLO	ID	1999.
UP	13336-SCOVILLE	ID	2055.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 16333-YUCCA MOUNTAIN, NV

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7260-HARRIMAN	TN	15.
NS	6979-DANVILLE	KY	177.
NS	7008-LOUISVILLE	KY	277.
NS	7009-JEFFERSONVILLE	IN	281.
NS	4797-MOUNT CARMEL	IL	406.
NS	4954-MOUNT VERNON	IL	469.
NS	4953-CENTRALIA	IL	491.
NS	10859-EAST ST LOUIS	IL	549.
NS	10858-ST LOUIS	MO	555.
NS	10494-CENTRALIA	MO	674.
NS	10498-MOBERLY	MO	697.
NS	10616-KANSAS CITY	MO	828.
UP	10616-KANSAS CITY	MO	828.
UP	10617-KANSAS CITY	KS	831.
UP	11823-LAWRENCE	KS	869.
UP	11697-TOPEKA	KS	899.
UP	11696-MENOKEN	KS	904.
UP	11681-MARYSVILLE	KS	979.
UP	11405-HASTINGS	NE	1089.
UP	11410-GIBBON	NE	1115.
UP	11352-NORTH PLATTE	NE	1193.
UP	11358-O FALLONS	NE	1242.
UP	13703-JULESBURG	CO	1310.
UP	13465-CHEYENNE	WY	1456.
UP	13462-LARAMIE	WY	1508.
UP	13494-GRANGER	WY	1784.
UP	13568-OGDEN	UT	1923.
UP	13595-SALT LAKE CITY	UT	1959.
UP	13630-LYNNDYL	UT	2071.
UP	14766-VALLEY	NV	2388.
USG	14766-VALLEY	NV	2388.
USG	16333-YUCCA MOUNTAIN	NV	2487.

ROUTE FROM: NS 15316-K-25, TN
TO: USG 15359-SRP, SC

RR	NODE	STATE	DIST
NS	15316-K-25	TN	0.
NS	7288-DOSSETT	TN	21.
NS	7286-KNOXVILLE	TN	46.
NS	7318-MORRISTOWN	TN	87.
NS	7343-ASHEVILLE	NC	167.
NS	7814-SPARTANBURG	SC	234.
CSXT	7814-SPARTANBURG	SC	234.
CSXT	7838-GREENWOOD	SC	300.
CSXT	7961-AUGUSTA	GA	371.
CSXT	7732-ROBBINS	SC	400.
CSXT	7717-DUNBARTON / WELL	SC	409.
USG	7717-DUNBARTON / WELL	SC	409.
USG	15359-SRP	SC	417.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix F
Description and Impacts of Storage Technology Alternatives

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

List of Tables

Appendix F
Description and Impacts of Storage Technology Alternatives

Summary

This appendix presents a description and evaluation of currently available spent nuclear fuel storage technologies, and their applicability to foreign research reactor spent nuclear fuel. These technologies represent the range of alternatives that would be available to implement the proposed action. Some of these technologies are currently in use at U.S. Department of Energy (DOE) facilities. Several dry storage cask and/or building designs have been licensed by the U.S. Nuclear Regulatory Commission (NRC) and are operational with commercial nuclear power plant spent fuel at several locations.

This appendix also discusses potential storage sites and impacts of foreign research reactor spent nuclear fuel storage at these locations. The major sections in this appendix are:

Section F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

Section F.2 Storage Technology Evaluation Methodology

Section F.3 Selection of Storage Technologies for Further Evaluation

Section F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

Section F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Section F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Section F.7 Economic Evaluation of Foreign Research Reactor Spent Nuclear Fuel Storage and Related Management Alternatives

Figure F-1 presents the different spent nuclear fuel storage technologies, which are divided into wet and dry systems and further classified by their materials of construction (i.e., concrete, metal), location (i.e., aboveground or belowground), and size (i.e., cask versus vault building or pool). The final level of detail is the specific design with 12 specific vendors' designs displayed in this figure. The following specific designs are of U.S. origin: Nuclear Assurance Corporation, MC-10, NUHOMS, and Ventilated Storage Cask-24 (Section F.1 describes these in more detail). The others are designed by foreign companies, but many of these companies, such as Transnuclear Inc., have U.S. affiliates. The principal categories of spent nuclear fuel storage technology are dry vault (building), dry cask, and wet pool.

This appendix discusses the aforementioned designs in terms of their shielding, criticality, thermal, structural, cost, and ease of use features. Some numerical design parameters are presented for comparison. Advantages and vulnerabilities of each design are also presented. Since none of these designs have been specifically designed or licensed for foreign research reactor spent nuclear fuel and related research reactor type fuel, some extrapolation has been made in this comparative assessment. All of the existing commercial designs and proposed new designs appear to be adaptable to foreign research reactor spent

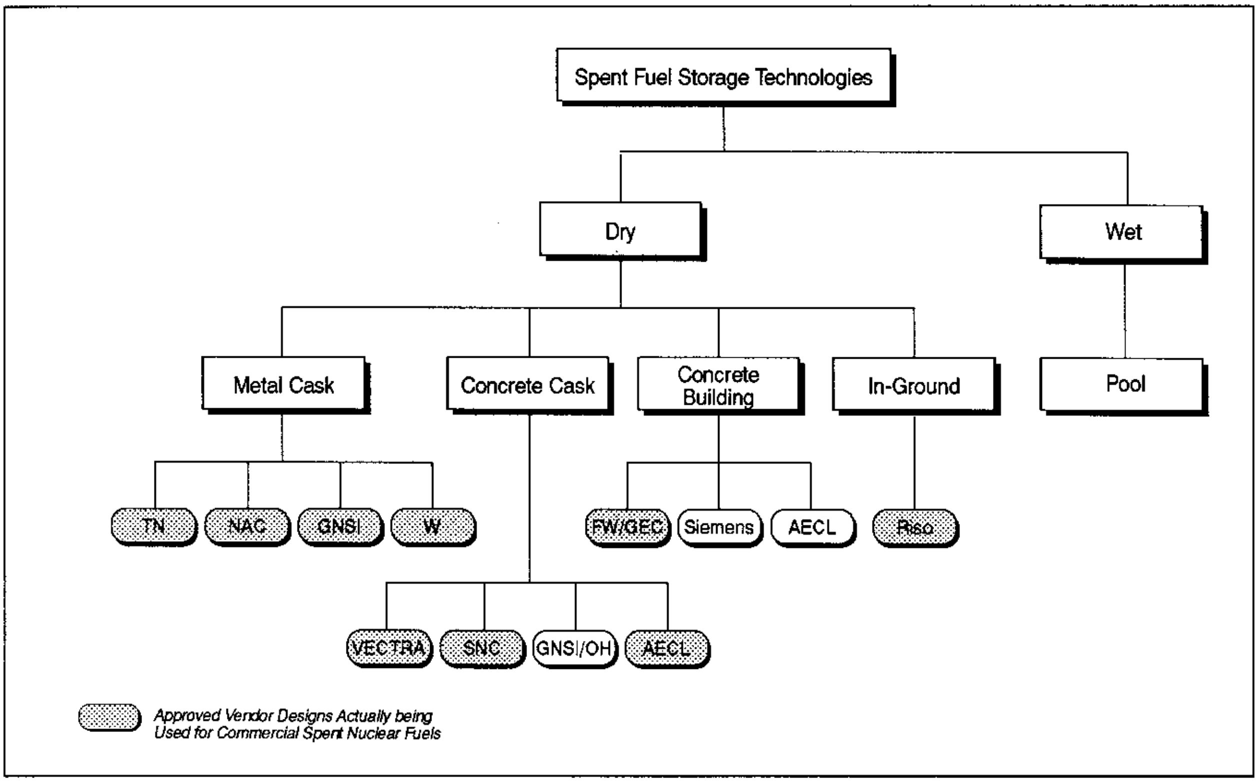


Figure F-1 Spent Nuclear Fuel Storage Technologies and Vendors

nuclear fuel with only minor, easily implemented modifications, such as interior baskets for holding the spent nuclear fuel. Use of existing facilities at a site for staging and characterization favors a cask storage approach, while a stand-alone, separate spent nuclear fuel storage approach requires a vault and other support facilities. Schedule and monetary considerations favor casks over the vault for sites with existing facilities, and this is why most domestic utilities are pursuing dry casks for long-term storage of spent nuclear fuel. Casks are the only independent spent nuclear fuel storage installation designs that have received certification by the NRC in accordance with 10 Code of Federal Regulations (CFR) 72 Appendix K.

The evaluation indicates that both wet and dry storage of foreign research reactor spent nuclear fuel appear acceptable for the time periods envisioned for the proposed action (i.e., through 2036). Commercial spent nuclear fuel dry storage systems require a minimum wet pool storage time or cooldown period of approximately 5 years after discharge from the nuclear reactor prior to emplacement into dry storage. In actual practice, this usually averages around an 8-year average cooldown period and, frequently, the commercial spent nuclear fuel has had over a 10-year cooldown period in a wet pool prior to emplacement into dry storage. This cooldown period ensures that licensed conditions for cladding temperatures (based upon potential corrosion, and usually around 350°C, or 630°F) are not exceeded. Foreign research reactor spent nuclear fuel has a lower cladding temperature limit based upon a phase transition in the aluminum metal cladding; this aluminum cladding limit has been identified as 175°C (315°F). Thus, a maximum cooldown period of 3 years of wet pool storage after irradiation has been identified for the foreign research reactor spent nuclear fuel prior to emplacement into dry storage. This would ensure that all foreign research reactor spent nuclear fuel elements were below a heat load of 40 Watts each, and most elements would be 10 Watts or less. The majority of the currently available foreign research reactor spent nuclear fuel already meets this requirement. Most of the existing dry storage designs appear acceptable for foreign research reactor spent nuclear fuel, without any clear preference. It should be noted that a research and development project to examine the applicability of aluminum-clad spent nuclear fuel dry storage at the Savannah River Site was initiated in Fiscal Year (FY) 1994.

The utilization of dry storage methods for foreign research reactor spent nuclear fuel requires the acquisition of racks, baskets, storage canisters, and/or casks. New construction would be required for dry vaults, except for several existing facilities at the Nevada Test Site and Idaho National Engineering Laboratory.

The utilization of wet storage methods requires a lined basin within a seismically qualified facility with the ability to maintain water chemistry and handle liquid radioactive waste. Currently, there are few existing DOE facilities in this category, and none have sufficient capacity to accommodate all of the foreign research reactor spent nuclear fuel. Thus, the selection of wet storage would require DOE acquisition of a facility, either by new construction or purchase of an existing facility such as the Barnwell Nuclear Fuels Plant (BNFP) that is owned by Allied General Nuclear Services. A summary of storage technology characteristics is given Table F-1. Sections F.4, F.5, and F.6 address environmental impacts, occupational dose, and accident consequences for storage. Section F.7 discusses costs in detail.

Table F-1 Summary of Storage Technology Characteristics for Commercial and Foreign Research Reactor Spent Nuclear Fuel

Storage Technology	*DOE Site Status (New or Existing)*	*Land Use Ha (Ac)*	*Annual Low-level Waste (m^3)*[a]	*Potential Annual Spent Nuclear Fuel Storage Public Impact (LCFs)*	*Lead Time until Spent Nuclear Fuel Storage (Years)*
Dry Vault - Utility Fuel	New	4 (5)	1-4	NA	2-3
Dry Cask (Concrete) - Utility Fuel	New	4 (5)	1-4	NA	2-3
Wet Pool - Utility Fuel	New	2 (3)	1-4	NA	3-5
Dry Vault - Savannah River Site Research Reactor Fuel	New	4 (8)	16	0	5-10[b]
Dry Cask - Savannah River Site Research Reactor Fuel	New	4 (5)	16	0	3-5[b]
Wet Pool - Idaho National Engineering Laboratory Research Reactor Fuel	New	2 (3)	12	2.4×10^{-12} to 2.5×10^{-10}	5-10[b]

NA = Not Available; LCF = Latent Cancer Fatality

[a] *Low-Level Waste generation decreases significantly if spent nuclear fuel is only being stored, without additional spent nuclear fuel receipts. To convert to ft^3, multiply by 35.3.*

[b] *To allow for extended periodic examination and characterization of fuel.*

DOE currently has pilot-scale experience with dry storage of spent nuclear fuel, and there are no identified technical constraints that would prevent dry storage of foreign research reactor spent nuclear fuel. There would be some need, however, for characterization, canning, and periodic inspection and monitoring. Both NRC-licensed and not yet licensed dry storage designs are readily available from commercial vendors. NRC-licensed designs have the following advantages:

- specific NRC requirements have been met that are equivalent to DOE requirements and guidance,
- extensive, interactive technical safety reviews have already been conducted between the supplier and the regulator,
- peer and public review has occurred as part of the licensing process,
- proven applications are in operation at commercial nuclear power plant sites, and
- documentation and quality assurance requirements have been satisfied.

For sites with an existing spent nuclear fuel infrastructure that includes facilities for spent nuclear fuel receipt, examination, and loading, a modular approach based upon casks can be implemented rapidly to meet Phase 1 requirements using standard funding and procurement capital appropriation methods. The casks could also be used for Phase 2, and their usage would avoid additional procurement. A modular dry vault approach represents an integrated self-contained, stand-alone facility, and can be used at any of the proposed management sites. However, construction of the vault could represent a major project or major systems acquisition under DOE management requirements, which may require a 7 to 10 year period for completion. Thus, a vault dry storage approach probably could not be available immediately. Metal cask development programs, such as dual- and multi-purpose casks, eliminate many storage site handling requirements and may provide future improvements.

Section F.7 evaluates the economics of the entire (40-year plus) foreign research reactor spent nuclear fuel program, including transportation, receipt/handling/inspection, storage, preparation for disposal, transportation to the repository, and disposal for the storage/disposal and chemical separation/vitrification alternatives. Costs are presented as rough-order-of-magnitude net present values, using a 4.9 percent real discount rate. In 1996 dollars, minimum total program costs for the storage alternative are about $800 million. This total divides into four very roughly equal parts: shipping to the United States and program management; receiving and storage at existing facilities; receiving, storage, and fuels qualification at not-yet existing facilities; and repository disposal. Other cost factors would be expected to add as much as $500 million to the program costs. Among the other cost factors are systems integration and logistics contingencies ($75 to $100 million), risks associated with limited characterization of the spent nuclear fuel ($100 million), risks associated with direct disposal of HEU ($50 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($200 million or more). Total costs, including all contingencies and risks could thus be in the $1.3 billion range.

For chemical separation alternatives, minimum total program costs are about $700 million. Savings in chemical separation and disposal of high-level waste versus storage and disposal of spent nuclear fuel account for the bulk of the difference between the costs in the chemical separation case and the storage case. Other cost factors would be expected to add as much as $250 million to the program costs. The key cost factors are systems integration and logistics contingencies ($75 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($100 million or more). If part of the material shipped to the Savannah River Site was chemically separated and part was stored, costs would typically be between the boundaries for all-separation and all-storage.

Hybrid alternatives that ship about 1/4 of the foreign research reactor spent nuclear fuel to the United Kingdom Atomic Energy Authority's Dounreay facility and manage the remainder as in the U.S. chemical separation case generate minimum total program costs of about $650 million. Other cost factors would be about the same as in the chemical separation case.

At the level of accuracy in the costs presented here, alternatives based on chemical separation of aluminum-based spent nuclear fuel in the United States are likely to cost about the same as alternatives that divert a significant fraction of the spent nuclear fuel (aluminum-based and TRIGA) to Dounreay. Alternatives based on storage and direct disposal of spent nuclear fuel or some non-separation processing approach (e.g., melt and dilute) are likely to cost several hundred million dollars more.

F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

In this section, two major generic technologies will be presented. International and domestic types of each technology will be addressed. Section F.1.1 will discuss the dry storage designs available. Section F.1.2 will address wet storage technology types. The range of alternatives available to each site for the implementation of the proposed action is presented in Section F.1.3.

F.1.1 Dry Storage Designs

F.1.1.1 Overview of Dry Storage Approaches

There are several types of dry storage technology currently in use or proposed by various vendors at DOE sites as well as at commercial nuclear power facilities. These include:

- aboveground free-standing metal casks,
- aboveground free-standing concrete casks,
- aboveground free-standing dry storage buildings (vaults),
- inground lined and unlined holes or wells with or without casks,
- hot cells (buildings), and
- aboveground free-standing multi-purpose or dual-purpose casks.

It should be noted that additional support facilities for transfer and staging operations may be required in order to use the aforementioned dry storage technologies. A short discussion of the advantages and disadvantages of all dry storage technologies is given in the following sections.

It is important to appreciate the different approaches to handling weight and shielding. Today, most spent nuclear fuel facilities utilize a wet pool environment for handling, storing, and transferring spent nuclear fuel to transportation casks. The pool water provides shielding [usually a 3-meter (m) or 10-ft water cover is the minimum requirement], confinement of contamination, decay heat removal, and thermal capacity. All spent nuclear fuel elements weigh less than 0.9 metric tons (1 ton) and are readily moved within the pool by a crane of that capacity. Transportation containers (casks) for highway transport weigh between 18 and 36 metric tons (20 and 40 tons), and rail casks can weigh up to 91 metric tons (100 tons). Thus, most wet pool facilities have a bridge crane spanning the storage areas and the receiving bay(s) with a capacity of 45 to 91 metric tons (50 to 100 tons). Economical dry storage requires that a large number of elements be stored in each cask. Cask weights exceeding 91 metric tons (100 tons) are possible.

Dry storage manufacturers have overcome this problem by using metallic "transfer" canisters. These transfer canisters are considerably lighter than transportation casks, and usually weigh in the 9 to 27 metric tons (10 to 30 ton) range before loading. The transfer canister provides some shielding, but is principally for confinement of the spent nuclear fuel. They are loaded in the same manner as transportation casks.

For dry cask storage, the canister is loaded onto a truck and transferred to a previously constructed, shielded concrete cask away from the wet pool. With a vault, the canister is moved by a crane within a concrete shielded facility and placed in a storage tube within concrete shielding.

Radioactive materials in spent nuclear fuel require two levels of confinement for dry storage. These are usually the cladding material and the metal container (or transfer canister) within the metal cask or concrete structure (cask or vault). Leaking fuel elements can be dry stored provided they are placed within a separate metal container (i.e., "can") within the canister. This is relatively easy to accomplish, but can consume additional storage space. For foreign research reactor spent nuclear fuel, the impacts of canned fuel upon storage capacity should be minimal. The amount of canning expected for foreign research reactor spent nuclear fuel is not yet determined.

F.1.1.1.1 Aboveground Free-Standing Metal Casks

Metal storage casks are generally robust and some may even have been originally designed to meet transportation requirements. They are resistant to seismic loads, high winds, design basis tornado missiles, and accidental drops. The mechanism for heat removal is simple, using direct metal conduction to the external surface which is cooled by natural convection. They are not subject to air pathway blockage by snow, ice, or flooding. The shielding is accomplished by various means, primarily thick steel, lead, or cast iron wall sections. The dry metal casks are passive, requiring minimal surveillance. There are no high-temperature thermal limits on cask material; however, if the material is cast iron or ferritic steel, there may be low-temperature thermal limits to prevent brittle fracture. Brittle fracture is a phenomenon that occurs in some materials such as glass at normal temperatures, or in cast iron or some steels (ferritic) at low temperatures. Fracture requires a stress to initiate. Thermal limits always apply to the fuel cladding. This type of dry storage has a proven track record in the United States and overseas.

The disadvantages of the metal cask designs are the following. Frequently, metal cask designs are more expensive than concrete/metal hybrid designs or dry vault storage designs. The current metal cask designs use dual compressible "O" rings with a pressure gauge to monitor the confinement seal. "O" rings are gaskets which, when compressed, form a gas-tight seal. Seal leakage is a possible event which must be considered for this design. The metal cask may be very heavy, thus imposing a limiting factor for cranes at existing facilities.

F.1.1.1.2 Aboveground Free-Standing Concrete Casks

The advantages of concrete casks, as compared to all other storage technologies, are given below. Concrete cask systems are inexpensive relative to metal casks. The concrete casks require no active systems because they are totally passive. They consist of a welded cylindrical container or basket enclosing the spent nuclear fuel which is then placed inside either a vertical or horizontal concrete structure. The concrete shielding structure may be fabricated onsite. This type of dry storage has been utilized at commercial nuclear power plant facilities, for example: H.B. Robinson, Oconee, Calvert Cliffs, and Palisades. Concrete casks have also been licensed for use at the Brunswick plant. Many other utilities are already committed to taking this route for the interim storage of their commercial spent nuclear fuel.

The disadvantages for concrete cask systems are: (1) more surveillance is needed than with metal casks to verify no blockage of air passages, (2) they are not licensed for transportation over public roads, (3) they require a special purpose onsite shielded transportation cask, and (4) the long-term concrete temperature limit restricts the heat load of the spent fuel. However, for the foreign research reactor spent nuclear fuel, heat loads and fuel cladding temperature limits are a small fraction of the commercial spent nuclear fuel values. Therefore, high concrete temperatures are expected to be avoided.

F.1.1.1.3 Aboveground Free-Standing Dry Storage Building (Vault)

Vault storage consists of a large concrete aboveground building enclosing a vertical or horizontal array of spent nuclear fuel storage metal tubes and support systems, The advantages for the vault type of dry storage, as compared to all other storage technologies, are the following. For large quantities of spent nuclear fuel assemblies, the vault may have economic advantages when compared with either type of cask system. The heat removal is passive. The heat removal capacity for a properly designed vault is large, and therefore, there should be little concern for thermal limits being imposed (although there may be individual fuel decay heat limits). The vault which is licensed in the United States and abroad, has no high temperature limit associated with concrete. However, there is a low temperature limit because the secondary fuel confinement barrier is ferritic steel. To comply with current NRC 10 CFR 72 regulations, all spent nuclear fuel storage systems must have two confinement barriers. The intact fuel cladding is considered the first confinement barrier, and the cask or vessel is considered the secondary confinement barrier. The vault has a major advantage over all other types of dry storage because it provides a shielded means for loading the spent nuclear fuel on the vault premises. Another important advantage of the vault is the ease of spent nuclear fuel retrieval and monitoring while in storage. The vault includes facilities for inspection, placement in containers, and drying of wet fuel. The weight/volume of stored fuel is not a limiting factor. This type of system is currently in use at Fort St. Vrain in Colorado, at Wylfa, Wales, and is under construction at the PAKs nuclear power plant in Hungary.

The disadvantage is that, for small quantities of spent nuclear fuel, the cost may be higher than either the metal or concrete cask systems since a vault requires greater capital outlays.

F.1.1.1.4 Inground Lined and Unlined Wells With or Without Casks

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced air heat removal is accomplished by directing the air around spent nuclear fuel containers and out through tubes embedded in the concrete. Like the pool storage systems, the RISO National Laboratory's system relies on active heat removal systems.

F.1.1.1.5 Hot Cell Facilities

Although hot cells are available at many facilities, including the Savannah River Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, they can best be considered for small quantities of spent nuclear fuel for very short periods of time. Hot cells are basically set up to perform various operations on hazardous materials, and are generally not spacious enough to store materials on an indefinite or long-term basis. Furthermore, hot cells are frequently contaminated; this contamination may pose problems when it is time to transfer the foreign research reactor spent nuclear fuel to the repository. It is important to note that DOE possesses several unique hot cells that may be capable of foreign research reactor spent nuclear fuel storage due to their large, vault-like design.

F.1.1.1.6 Aboveground Free-Standing Multi-Purpose or Dual-Purpose Casks

The dual-purpose cask combines the functions of interim storage at a designated site and transportation on public roads, rail systems, or waterways. A multi-purpose cask may also add a third function of a repository canister; i.e., the cask and its contents need no further processing, characterization or identification in order to be compatible with the final repository. A dual-purpose or multi-purpose cask has attractive possibilities for the storage of spent nuclear fuel, regardless of the type of reactor (i.e., commercial or research reactor). Dual-purpose designs would satisfy the two functions of storage and transportation. For commercial utilities, this implies satisfaction of 10 CFR 71 requirements for transportation and 10 CFR 72 requirements for storage. By minimizing fuel handling operations, the dose for workers can be reduced, and the number of additional low-level waste products can be reduced. Minimization of fuel handling may also result in cost reductions, although this case has not been made. For a multi-purpose cask, satisfaction of 10 CFR 60 requirements is also necessary. DOE's Office of Civilian Radioactive Waste Management was actively pursuing a program to develop multi-purpose canister for domestic use (EG&G, 1994b; DOE, 1994f; DOE, 1994b; DOE, 1994c). However, DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this project.

The dual-purpose cask systems that are currently proposed offer a reduction in handling of the spent nuclear fuel in the storage to transportation operations, and multi-purpose cask systems offer an even greater reduction in handling in the operations involved at the repository site. However, no detailed cost/benefit analyses have been undertaken for foreign research reactor spent nuclear fuel. Furthermore, there is no basis at this time for concluding that either the "waste form" (intact spent nuclear fuel assemblies) or the sealed container will be compatible with the repository requirements. It is premature to draw any conclusion on the desirability to proceed with a multi-purpose cask system for foreign research reactor spent nuclear fuel use.

F.1.1.2 Specific Dry Storage Designs

There are no currently licensed dry storage systems specifically for foreign research reactor spent nuclear fuel in the United States. There are, however, many examples of dry spent nuclear fuel systems licensed by the NRC for commercial fuel. Table F-2 provides an overview of current manufacturers of dry storage systems. Table F-3 is a listing of dry storage systems currently licensed in the United States.

Dry storage systems must meet many design criteria, such as protection of fuel from degradation, shielding, thermal, criticality safety, structural integrity of confinement vessel, structural integrity of shielding, mechanical handling of fuel assemblies or canisters, containment and operational aspects. Some of these criteria are interrelated. For example, thermal criteria are designed to maintain fuel and cladding structural integrity. Shielding, thermal, and criticality parameters are the most important and are discussed in the following sections.

F.1.1.2.1 Shielding Design Comparisons

A spent nuclear fuel storage system must provide for adequate shielding of both the gamma and neutron radiation that emanate from irradiated nuclear fuel. The shielding must be designed to reduce the combined gamma and neutron dose rate to values that are below the limits for the public at the site boundary, collocated workers, and workers at the fuel storage facility. These limits are determined by Federal regulations such as 10 CFR 72 and 10 CFR 20. Shielding is designed for the maximum expected gamma and neutron source term, which is determined by performing computer code analyses of the nuclear fuel that account for the initial fuel fissile material inventory, its burnup in the reactor core, and the time after removal from the reactor (i.e., decay time) prior to its anticipated placement in the storage facility. The selection of a bounding and conservative set of these parameters results in the calculation of the highest possible gamma and neutron source term to be used in shielding design and analyses.

Shielding for gamma radiation relies on the use of high atomic weight or density materials, which attenuate and absorb gamma rays. The material selection depends on design limitations regarding shield thickness, cost, strength, and weight. The five materials which are almost always used in spent nuclear fuel storage facilities for gamma shielding are water, lead, steel, ductile iron or concrete. Lead and steel, having much higher densities and atomic weights than concrete and water, can provide relatively more

Table F-2 Dry Storage Technology Systems

Company	*Metal Cask*	*Concrete Cask*	*Building (Vault)*
AECL/Transnuclear	---	SILO	MACSTOR/CANSTOR
FW/GEC	---	---	MDV
GNS/GNSI	CASTOR	---	---
GNSI/OH	---	HDC	---
Nuclear Assurance Corp.	NAC	---	---
VECTRA	---	NUHOMS	---
Sierra Nuclear Corporation	---	VSC	---
RISO National Laboratory	---	DR3	---
Siemens Power Corporation	---	---	FUELSTOR
Transnuclear, Inc.	TN	---	---
Westinghouse Electric Corporation	MC-10	---	---
Atomic Energy of Canada, Ltd.	---	SILO	CANSTOR
Total Design:	**4**	**6**	**4**

FUELSTOR = Fuel Encapsulation and Lag Storage; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.; GNSI/OH = GNSI of Ontario Hydro; VSC = Ventilated Storage Cask; MDV = Modular Dry Vault

Table F-3 Dry Storage Systems Currently Licensed in the United States

Manufacturer	*System*	*Location*
General Nuclear Systems, Inc. (Chem-Nuclear)	CASTOR V/21	Surry
VECTRA	NUHOMS-7P, NUHOMS-24P, & NUHOMS-52B	Robinson, Oconee, & Calvert Cliffs
Westinghouse Electric Corporation	MC-10	Surry
Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)	MDV	Wylfa, Wales UK PAKS (Hungary) Fort St. Vrain (USA)
Transnuclear	TN-24 & TN-40	Surry, Prairie Island
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Palisades
Nuclear Assurance Corporation	NAC-C28, NAC-I28, NAC S/T, & NAC S/TC	Surry

effective gamma shielding with a smaller thickness of material. However, using steel and/or lead imposes a design penalty of increased cost. Concrete and water are much less expensive and may reduce overall shielding costs. It should also be noted that lead can be categorized as a Resource Conservation and Recovery Act waste, which restricts future decommissioning and disposal options. Concrete and water may also present unique safety problems, such as leakage (for water) or cracking (for concrete) during postulated accidents.

Neutron shielding requires low atomic weight materials because the uncharged neutron can only be absorbed by reducing its energy in collisions with nuclei similar in mass. Since the mass of a neutron is approximately one atomic mass unit, low atomic mass elements such as hydrogen, inert gas, lithium, carbon, and boron are suitable shields. Hydrogenous materials such as concrete and water are typically used in neutron shielding. It should be noted, however, that a sufficient thickness of heavier materials (such as high carbon steels) can provide neutron shielding. Also, shielding manufacturers offer products that have been artificially fortified in their hydrogen content, such as special forms of concrete, borated resin, and polyethylene. As in the case for gamma shielding, design factors in material selection include cost, density, weight, and safety.

The design of spent nuclear fuel storage facility shielding must also incorporate other factors along with cost, density, weight, and safety. Shielding usually performs a second function as a heat transfer medium from the spent nuclear fuel decay heat to the environment, and must therefore be able to effectively remove heat without exceeding fuel and shielding storage temperature limits. In some instances, the shielding also performs a structural function, either in handling or support.

Table F-4 shows a comparison of specific designs with a view toward shielding considerations. All of these designs will be discussed in more detail in subsequent sections of this appendix.

F.1.1.2.2 Thermal Design Comparisons

Spent nuclear fuel storage facilities are designed to effectively remove spent nuclear fuel decay heat during both incident-free operation and postulated accident conditions. Thermal design limits include long-term fuel storage cladding temperature to maintain cladding integrity and, in some cases, temperature limits of structural and/or shielding materials. Unlike pool storage systems, most of the dry storage systems emphasize passive heat removal. In contrast, active systems in wet pools include pumps, make-up water systems, filtration and water treatment systems, and heat exchangers.

All dry storage designs encapsulate the fuel, after it is dried, in a metal canister or tube that is evacuated (vacuum dried) and then filled with an inert gas such as helium. Helium is frequently used for its relatively high thermal conductivity that enhances heat conduction and heat transfer from the fuel to the encapsulating metal canister. Helium's inert properties also inhibit cladding corrosion. Since all the dry fuel storage technologies utilize a metal canister to enclose spent nuclear fuel, the first modes of heat transfer from the fuel to this canister's walls are heat conduction and radiation from the fuel cladding surface through the inert gas to the inside wall of the metal canister. Decay heat transfer from the encapsulating canister to the environment is accomplished by several different mechanisms dependent upon the specific storage design technology.

The dry metal cask design relies on its solid thick metal wall for conduction heat transfer from the fuel storage cavity to the atmosphere. Metal cask conduction heat transfer is not susceptible to any accident or degradation. This thermal design is inherently easy to analyze because conduction is a well-known heat transfer mechanism, and the thermal conductivity of such metal cask materials as carbon steel, stainless steel, and ductile cast iron is well known over the range of temperatures and conditions that are expected in the cask while storing spent nuclear fuel. With known design fuel decay heat, cask geometry (i.e., cask wall thickness), conduction material composition, and suitably conservative heat transfer assumptions from the cask metal surface to the ambient air, the temperature distribution within the cask and maximum fuel cladding temperature can be calculated with a high degree of certainty.

The dry concrete cask design uses a combination of conduction, natural convection, and radiation heat transfer to remove decay heat from the stored spent nuclear fuel and maintain acceptable operating temperatures. An air passageway around the storage canister is provided in this design because the relatively low thermal conductivity and allowable operating temperature limit of concrete, as compared to metal, prevent the concrete shield walls from serving as the primary means of decay heat removal. Radiation streaming requires that the inlet and outlet air passages to the cavity surrounding the canister be designed as a geometric labyrinth with suitable bends. One concrete cask design, the Atomic Energy of Canada, Ltd. SILO, does not have air passages but instead relies solely on conduction through solid

Table F-4 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	Model	Shield Material	Shield Thickness	Design Limit Surface Dose Rate[a]
Nuclear Assurance Corporation	S/T	S.S., Lead, NS4FR	Radial: 20.3 cm (8 in) S.S. & 17.8 cm (7 in) NS4FR Axial: 12.7 cm (5 in) S.S.-PB & 7.6 cm (3 in) NS4FR	1 milliSievert/hr (100 mrem/hr)
Transnuclear, Inc.	TN-24	Borated Resin, C.S.	NA	Side: 0.57 milliSievert/hr (57 mrem/hr) Top: 0.11 milliSievert/hr (11 mrem/hr) Bottom: 0.45 milliSievert/hr (45 mrem/hr)
	TN-40	Borated Resin, C.S.	Radial: 21.6 cm (8.5 in) C.S. 11.4 cm (4.5 in) Resin Bottom: 22.2 cm (8.75 in) C.S. Top: 15.9 cm (6.25 in) Cast Iron	Side: 0.58 milliSievert/hr (58 mrem/hr) Top: 0.26 milliSievert/hr (26 mrem/hr) Bottom: 12.75 milliSievert/hr (1,275 mrem/hr)
Westinghouse Electric Corporation	MC-10	NS-3, C.S.	Radial: 25.4 cm (10 in) Steel, 7.6 cm (3 in) NS-3 Bottom: 25.4 cm (10 in) steel	2 milliSievert/hr (200 mrem/hr)
General Nuclear Systems, Inc.	CASTOR V21	Cast Iron, S.S., Polyethylene Rods	Radial: 30.5 cm (12 in) Bottom: 27.9 cm (11 in) Top: 39.1 cm (15.4 in) Rods Radial: 72-6.1 cm (2.4 in) Diameter	2 milliSievert/hr (200 mrem/hr)
VECTRA	NUHOMS 7P, 24P, and 52B	Concrete, S.S.	Side: 45.7/60.1 cm (18/24 in) Rear: 60.1 cm (24 in) Roof: 91.4 cm (36 in)	2 milliSievert/hr (200 mrem/hr) (at air inlet)
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Concrete RX-277, Hydrogenated Concrete, C.S.	Radial: Steel & Concrete Top: RX-277 & Steel Bottom: Steel & Concrete	Side: 0.20 milliSievert/hr (20 mrem/hr) Top: 0.50 milliSievert/hr (50 mrem/hr) Air Inlet or Outlet: 1 milliSievert/hr (100 mrem/hr)
FW/GEC	MDV	Concrete	106.7 cm (42 in)	0.21 milliSievert/hr (21 mrem/hr)

NA = Not Available; C.S. = Carbon Steel; Pb = Lead; S.S. = Stainless Steel; NS-3 = Concrete; NS4FR = Special Fire-Resistant Castable Resin; RX-277 = Special Concrete with Extra Hydrogen; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); MDV = Modular Dry Vault

[a] *These are limits established for commercial spent nuclear fuel assemblies. The dose rate expected from storage of foreign research reactor spent nuclear fuel is likely to be lower.*

concrete. The SILO's thermal design is acceptable only because it is limited to a much smaller total decay heat power than the air passage concrete casks.

Since a passive design is an underlying requirement of all dry concrete cask designs, the total airflow path from the cask air inlet to its outlet must include a sufficient elevation change to ensure natural convection airflow under all expected meteorological and heat load conditions.

The heat transfer from the canister follows two parallel paths: (1) convection from the surface of the canister to the naturally-induced airflow through the canister cavity, and (2) radiation and conduction heat transfer from the canister across the air in the cavity to the concrete shield and then conduction through the concrete shield wall thickness to the ambient air outside the concrete. Natural convection air heat removal is greater than the radiation and conduction through the air layer and concrete shield.

The heat transfer design of the concrete cask is vulnerable to accidents in which significant blockage of the air inlets and/or outlets restricts or prevents sufficient airflow into the canister cavity. Multiple inlets and outlets at different, and sometimes diametrically opposed, locations around the cask are used to reduce the likelihood of such an accident. Conservative adiabatic heatup analyses for these designs with commercial spent nuclear fuel have shown that temperature limits are not approached in more than 24 hours, even if the airflow inlets and outlets are completely blocked. Therefore, concrete cask sites have included a daily visual surveillance frequency for inspection of air inlets and outlets to ensure that they are not blocked. The adiabatic heatup for foreign research reactor spent nuclear fuel and its concomitant surveillance frequency may be different.

The concrete cask thermal design also requires more complex analyses for temperature distribution in both the fuel and the concrete due to the complex multidimensional and combined conduction-radiation-convection modes of heat transfer. An important thermal design issue for the concrete casks is proof that natural convection buoyancy-driven airflow will be induced through the inlet-cavity-outlet path under the entire range of expected wind and decay heat conditions, including the possibility of partial blockage that may be obscured from outside visual inspections. Unlike metal casks, which only have the fuel cladding temperature as a thermal limit, concrete casks are also limited by both the absolute magnitude and gradients of temperature within the concrete.

The concrete vault storage building represents a larger version of the concrete cask design in the realm of heat transfer. An array of vertically or horizontally oriented metal canisters enclosing spent nuclear fuel is surrounded by a concrete building with labyrinth air inlet and outlet passages. With the exception of size, this design utilizes the same modes of heat transfer as the concrete cask. Its inherently larger flow areas for inlets and outlets and typically larger elevation from inlet to outlet provide a greater natural convection airflow and reduce vulnerability to airflow passage blockage.

Specific Thermal Features

Thermal design performance parameters of specific manufacturers' dry storage technologies are presented in Table F-5. This table shows that all dry spent nuclear fuel storage technologies use radiation and conduction as heat transfer mechanisms, and that concrete-based systems also rely on internal air passage natural convection heat transfer. All the systems have fuel cladding temperature limits, but systems relying on concrete also have concrete temperature limits.

Pool storage systems utilize an active cooling system with pumps and heat exchangers that remove decay heat transferred to the pool water from stored fuel via conduction and natural convection. The relatively large mass and heat capacity of the pool water provide a significant margin of time before the pool water reaches its boiling temperature in the event of a cooling system failure.

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced-air heat removal is accomplished by directing the heating, ventilation, and air conditioning air around the stored fuel, and then out through separate tubes embedded in the concrete. Like the pool, the RISO National Laboratory's system relies on active heat removal systems.

The Atomic Energy of Canada, Ltd., SILO is an exception to the previously discussed concrete cask designs because it relies solely on conduction through a solid concrete structure for decay heat removal,

Table F-5 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	Model	Design Heat Load (KW)	Thermal Limits[a] (°C)	Heat Transfer Mode(s)[b]
Nuclear Assurance Corp.	S/T	26	NA	Conduction, Radiation
	S/TC	22	NA	Conduction, Radiation
Transnuclear, Inc.	TN 24	24	149 - Resin	Conduction, Radiation
	TN 40		NA	Conduction, Radiation
Westinghouse Electric Corp.	MC-10	13.5	340 - LWR Cladding	Conduction, Radiation
GNSI	CASTOR V21	21	370 - LWR Cladding	Conduction, Radiation
VECTRA	NUHOMS 7P and 24P	24	340 - Fuel Clad Normal 570 - Fuel Clad Accident	Conduction, Radiation, Natural Convection
	Standardized 24P and 52B	24 and 19	225 - Concrete Accident 177 - Concrete Normal 578 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
Sierra Nuclear Corporation	Ventilated Storage Cask-24	24	93 - Concrete Normal 177 - Concrete Accident 570 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
FW/GEC	Modular Dry Vault	0.15 per HTGR[c] Canister	399 - for Fort St. Vrain Type Fuel	Conduction, Radiation, Natural Convection

NA = Not Available; LWR = Light Water Reactor; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.

[a] *Fuel Limits are for Commercial Light Water Reactor Zircaloy Clad Fuel Type or for HTGR fuel.*

[b] *Heat transfer modes are for commercial spent nuclear fuel.*

[c] *HTGR= High Temperature Gas Reactor Type Fuel from Fort St. Vrain.*

without internal natural convection airflow around the canister. The SILO can maintain acceptable concrete and fuel cladding temperatures without internal airflow passages by limiting its contained total fuel heat load to about 4 kilowatts, as compared to the 24 kilowatts typical of other concrete casks with airflow passages. This lower heat load may not be limiting for the storage of foreign research reactor spent nuclear fuel since it does not produce decay heat as high as that for commercial nuclear power plant fuel for the same decay time.

F.1.1.2.3 Criticality Prevention Design of Spent Nuclear Fuel Storage Technology

A self-sustaining nuclear fission process is called criticality. Unlike the previously discussed thermal and shielding designs, criticality prevention design for spent nuclear fuel storage facilities does not rely on materials outside of the fuel storage basket or canister. Instead, the canister interior fuel support structure and fuel specifications for storage are the determining factors in criticality control.

Spent nuclear fuel storage facilities are shown to meet specific regulatory subcriticality requirements by conservative criticality analyses. These analyses conservatively assume that the spent nuclear fuel has its original enrichment of fissile material [e.g., fresh unirradiated fuel weight percent uranium-235 (^{235}U)]. In reality, the fuel has been irradiated and the initial concentration of fissile material is reduced from its original value through fission reactions producing numerous fission products.

Suitable criteria for establishing nuclear criticality safety have been documented (ANSI, 1984b, 1983, and 1975b). These documents deal specifically with, respectively, the storage of commercial spent nuclear fuel outside of the reactor and in dry storage installations.

Another conservative aspect of these criticality analyses is the requirement that a sensitivity study be performed that varies the water concentration within the canister free volume from no water to 100 percent water, to optimize moderation density. These analyses usually show that the most reactive (i.e., closest to critical conditions) configuration occurs with a water density less than that equivalent to a fully flooded canister (related to enrichment).

The criticality analyses explicitly model the fuel geometry, all materials present in the fuel, and the structural spacer design within the canister. Center-to-center distance for the fuel in the canister is another important parameter in determining the reactivity of the stored fuel.

In summary, the criticality prevention design of spent nuclear fuel storage facilities ensures that each canister will remain subcritical throughout the entire operation, during both incident-free and accident conditions. The criticality prevention design incorporates the following features:

- fuel specifications, including type of fuel, maximum initial fresh fuel ^{235}U enrichment, and number of fuel assemblies to be stored in a single canister,
- fuel assembly spacing inside the canister as set by the presence of structural support and spacing members, and
- the presence and composition of any neutron absorbing material between adjacent fuel assemblies inside the canister.

F.1.1.2.4 Current NRC-Licensed, Dry Storage Technologies for Commercial Spent Nuclear Fuel

The technologies discussed in this section are described in terms of their use for storage of commercial spent nuclear fuel. Foreign research reactor spent nuclear fuel storage design parameters will be different for each technology.

F.1.1.2.4.1 Nuclear Assurance Corporation S/T, NAC-C28 S/T, NAC-I28

Description of Nuclear Assurance Corporation S/T, NAC-C28 S/T and NAC-I28

Two of the Nuclear Assurance Corporation metal casks for the storage of spent nuclear fuel are in use at the Surry Nuclear Power Plant in Virginia. The Nuclear Assurance Corporation S/T design uses a combination of stainless steel and lead for gamma shielding and NS4FR, which is a fire-resistant castable resin, for neutron shielding (NRC, 1988a). To ensure a surface contact dose rate of less than 100 mrem/hr, 20.3 cm (8 in) of stainless steel and lead and 17.8 cm (7 in) of NS4FR are used in the cylindrical wall, while the top and bottom shields are composed of 7.6 cm (3 in) of NS4FR and about 12 cm (5 in) of steel and lead. Total weight of the loaded cask is either 91 metric tons (100 tons) for 26 intact Pressurized Water Reactor fuel assemblies, or 112 metric tons (124 tons), which is just under the 125-ton limit of many loading cranes for the 56 consolidated fuel assembly model. The Nuclear Assurance Corporation S/T models have been licensed by the NRC. The Nuclear Assurance Corporation S/T is shown in Figure F-2.

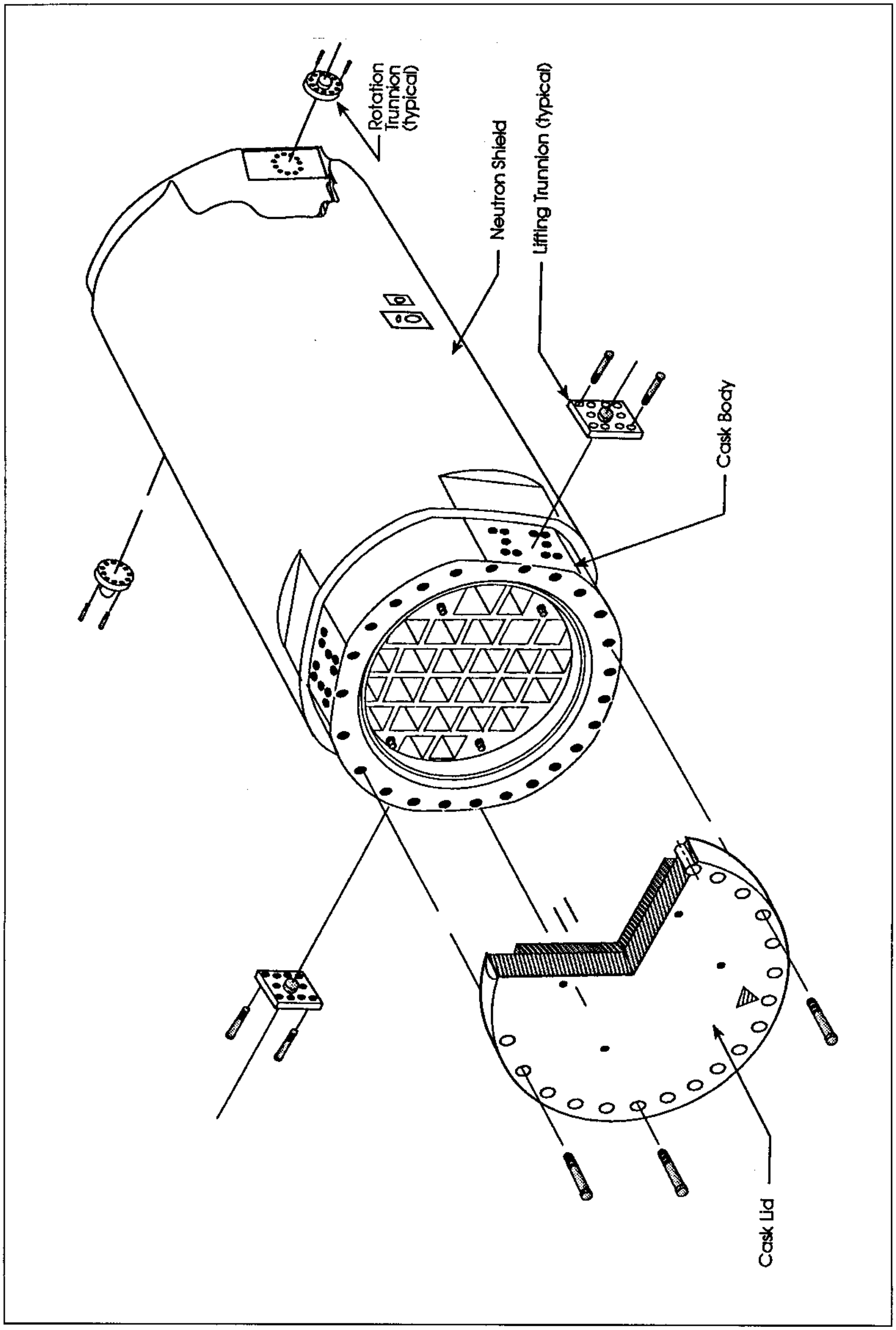


Figure F-2 Nuclear Assurance Corporation Spent Nuclear Fuel Storage Cask Body NAC S/T

NRC Certification or Basis for License

The NRC has granted a Certificate of Compliance to Model Nuclear Assurance Corporation S/T (Certificate Number 1002). Nuclear Assurance Corporation Model NAC-C28 S/T is also certified with Certificate Number 1003. The basis for these certificates is 10 CFR 72 Subparts K and L. Nuclear Assurance Corporation model NAC-I28 is currently licensed on a site-specific basis at Surry Nuclear Power Plant based on 10 CFR 72 Subparts A through I.

F.1.1.2.4.2 General Nuclear Systems, Inc. CASTOR V/21

Description of General Nuclear Systems, Inc. CASTOR V/21

In the United States, the General Nuclear Systems, Inc. CASTOR V/21 has been approved by the NRC and is in use at the Surry Nuclear Power Plant. This design relies on thick ductile cast iron and polyethylene as both its gamma and neutron shields. Ductile cast iron contains significant quantities of nodular graphite, which is essentially carbon, a good neutron shield. Polyethylene is a form of plastic that is high in hydrogen. The ductile cast iron shield is 30.5 cm (12 in) thick. Additional neutron shielding is provided by seventy-two 6.1 cm (2.4 in) diameter polyethylene rods placed in axial holes in the cast iron wall. The top lid shielding is 39.1 cm (15.4 in) of stainless steel, and the bottom lid shielding is 27.9 cm (11 in) of ductile cast iron. The V/21, holding 21 Pressurized Water Reactor fuel assemblies at Surry, weighs 96 metric tons (106 tons) fully loaded. A sketch of the CASTOR V/21 is presented in Figure F-3. The shielding design basis is for a surface contact dose rate less than 200 mrem/hr. There is a wide range of CASTOR designs for a variety of fuel types, including test reactor fuel. A conceptual design [CASTOR Material Test Reactor (MTR) 2] for a dual-purpose, transport/storage cask for research reactor fuel has been developed. This cask uses the same basic ductile cast iron body for shielding.

NRC Certification or Basis for License

The NRC has granted Certificate of Compliance Number 1000 for the General Nuclear Systems, Inc. model CASTOR V/21 under the terms of 10 CFR 72 Subparts L and K (Models X/28 and X/33 are not currently licensed, but are being reviewed by the NRC).

F.1.1.2.4.3 Westinghouse Electric Corporation MC-10

Description of Westinghouse Electric Corporation MC-10

The Westinghouse Electric Corporation MC-10 metal cask has been approved by the NRC and is in use at the Surry Nuclear Power Plant site (NRC, 1987). This cask design utilizes thick carbon steel and BISCO NS-3 hydrogenated concrete for shielding. The NS-3 provides neutron shielding, while the carbon steel is used for gamma shielding. Total radial neutron and gamma shielding is approximately 33 cm (13 in), while axial shielding is about 25.4 cm (10 in). The design surface contact dose rate is 200 mrem/hr, which bounds the actual vendor-calculated maximum surface contact dose rates of 7, 38, and 57 mrem/hr at the top, side, and bottom of the cask. The MC-10 was designed to hold 24 Pressurized Water Reactor fuel assemblies and weighs 103 metric tons (113.3 tons) fully loaded.

NRC Certification or Basis for License

The NRC has issued Certificate of Compliance Number 1001 for the metal cask model MC-10 in accordance with the terms of 10 CFR 72 Subparts L and K.

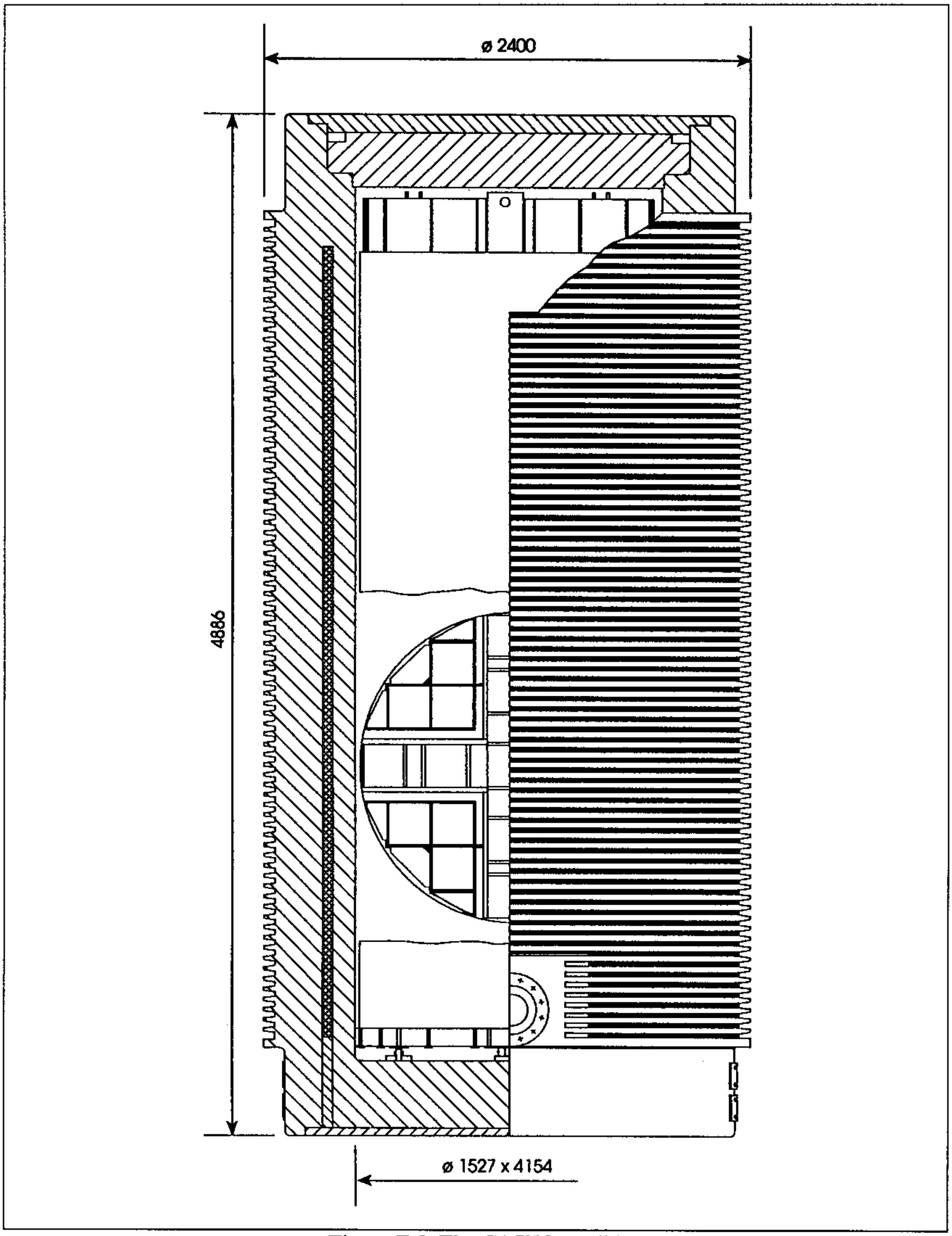


Figure F-3 The CASTOR V/21

F.1.1.2.4.4 Transnuclear, Inc. TN-24 and TN-40

Description of Transnuclear, Inc. TN-24, TN-40

The Transnuclear, Inc. design has been developed and produced for a large number of storage and transportation systems for radioactive materials, including spent nuclear fuel. The TN-24 and TN-40 models store 24 and 40 spent Pressurized Water Reactor fuel assemblies, respectively. TN systems feature metal casks for both transportation and storage of spent fuel. The TN-24 is an NRC-licensed storage cask that uses carbon steel for gamma shielding and a borated resin for neutron shielding (NRC, 1989). The TN-40 is a newer model that uses a two-metal shell design, with the inner shell consisting of high quality carbon steel for containment and the outer shell providing shielding and heat transfer, but of a lower quality steel. For the two models, top and side contact dose rate limits are less than 100 mrem per hour, but the bottom of the cask may have a contact dose rate limit as high as 1,275 mrem/hr. It should be noted that the normal configuration for these casks is to be standing upright on their bottoms, thereby precluding exposure to this relatively higher dose rate. A sketch of a Transnuclear, Inc. TN cask is shown in Figure F-4.

NRC Certification or Basis for License

The TN-24 model has been issued NRC Certificate of Compliance Number 1005 and is licensed according to 10 CFR 72 Subparts L and K. The TN-40 model is licensed on a site-specific basis at the Prairie Island Nuclear Power Plant in Minnesota (owned by Northern States Power) under the provisions of 10 CFR 72 Subparts A through I. The Transnuclear, Inc. Model TN-32 is not yet approved.

F.1.1.2.4.5 VECTRA Design NUHOMS-7P, -24P, and -52B

Description of VECTRA NUHOMS-7P, -24P, and -52B

VECTRA's NUHOMS designs utilize a horizontal concrete dry storage system for spent nuclear fuel (NUTECH, 1988). The NUHOMS-7P and NUHOMS-24P designs have been approved by the NRC for Pressurized Water Reactor spent nuclear fuel and are in use at the Robinson, Oconee, and Calvert Cliffs Nuclear Power Plant sites. The NRC approved the use of NUHOMS-52B for the Brunswick power plant, but the utility shipped this spent nuclear fuel to its Robinson plant. The NUHOMS design uses concrete as both gamma and neutron shielding. The requirement for internal air passages to allow natural convection heat removal from the metal storage canister placed within the concrete structure required 90 degree bends in the concrete shield for air passages to avoid radiation streaming and more detailed shielding analyses. The reinforced side wall concrete shield thickness is 45.7 or 61 cm (18 or 24 in) depending on location in the array, while the rear wall is 61 cm (24 in) thick and the roof is 91.4 cm (36 in) thick. The maximum surface contact dose rate limit at the air inlet is 200 mrem/hr. A sketch of the NUHOMS-24P system is shown in Figure F-5.

NRC Certification or Basis for License

NUHOMS models 7P and 24P are licensed at specific sites under the provisions of 10 CFR 72. Vectra has also received a license from the NRC for their standardized NUHOMS-24P and -52B models for use by the light water reactor utilities.

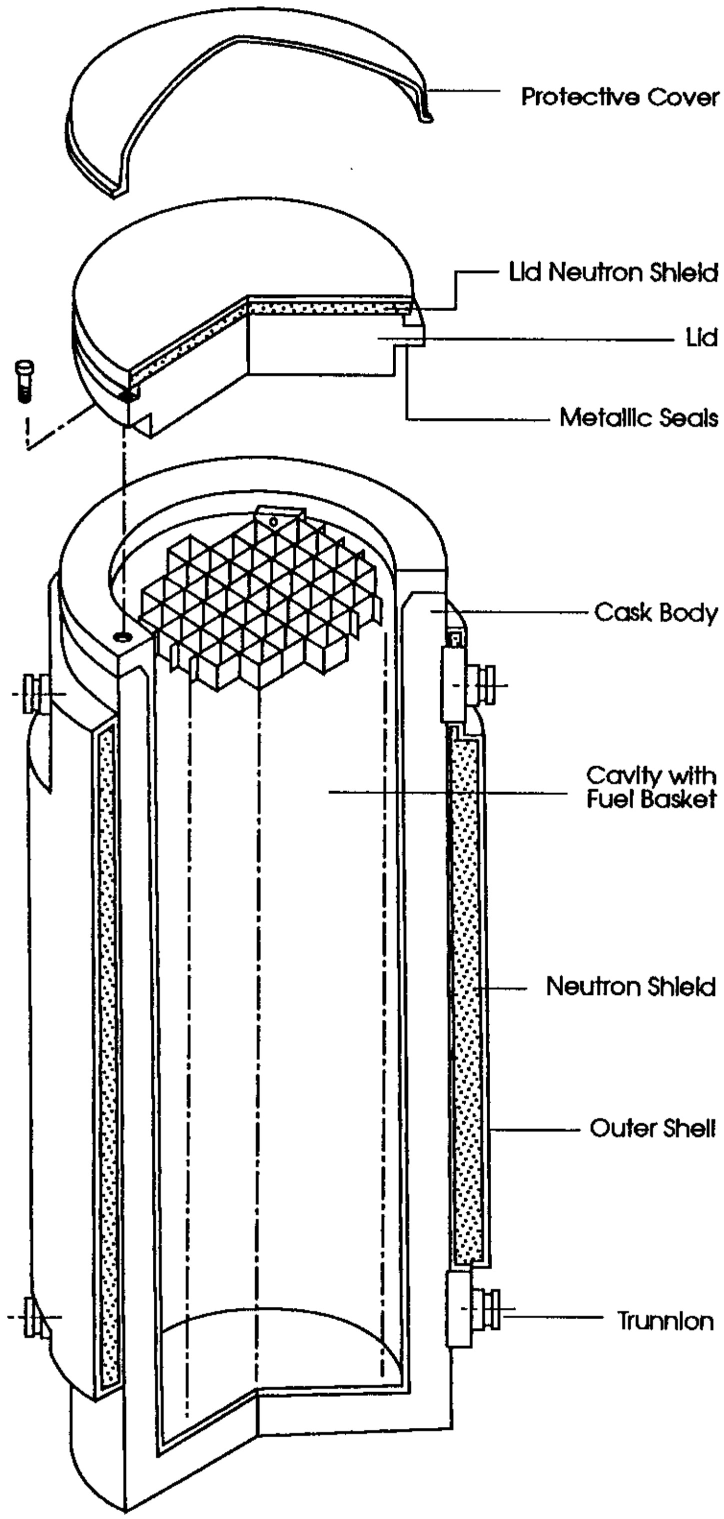


Figure F-4 The Transnuclear, Inc. TN Cask

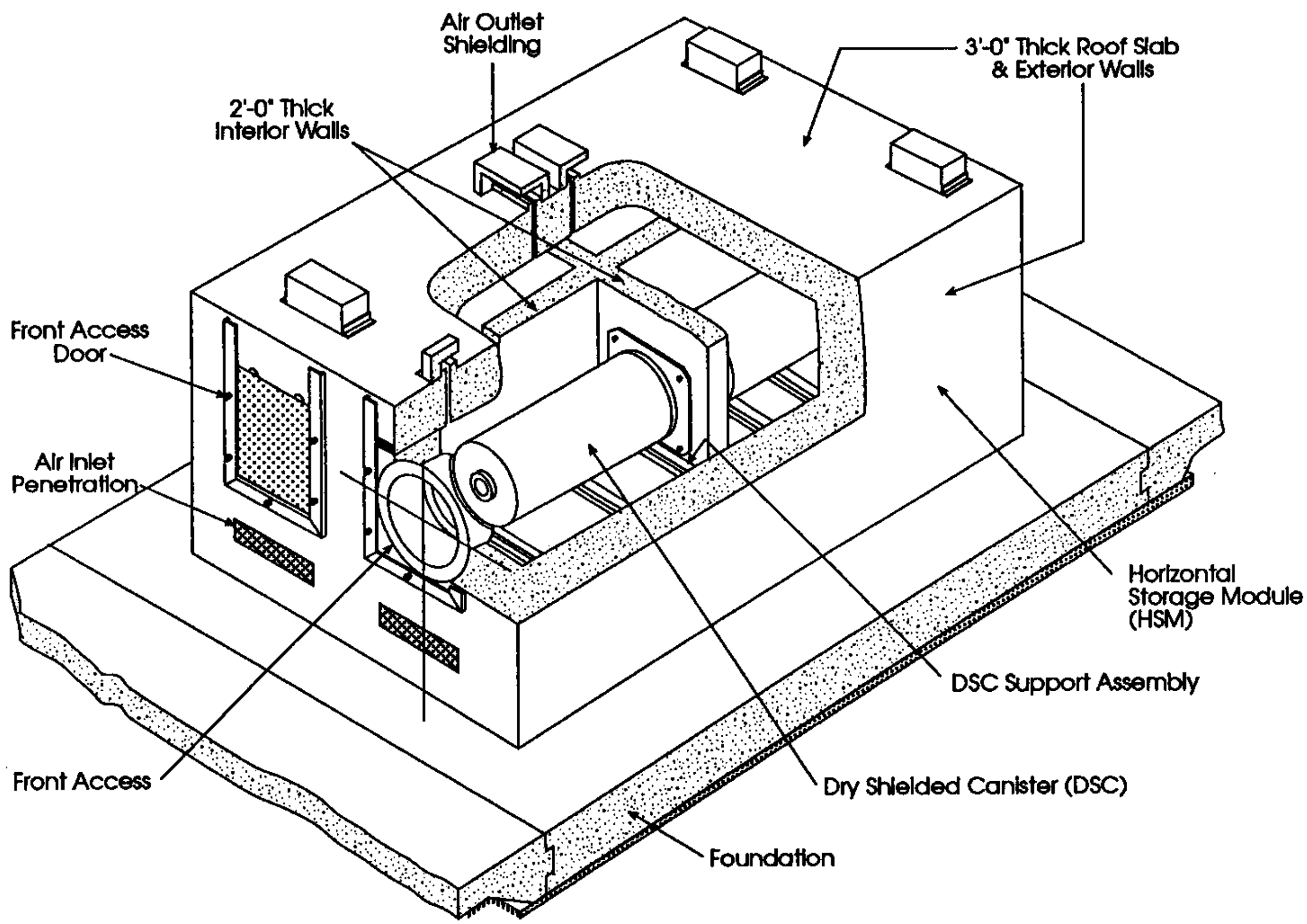


Figure F-5 The NUHOMS-24P Horizontal Storage Module Components

F.1.1.2.4.6 Modular Dry Vault

Description of Modular Dry Vault

The modular dry vault spent nuclear fuel storage system [designed by Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)] is the only vault system in the United States that has been approved by the NRC and is in operation at the Fort St. Vrain nuclear power plant site. The modular dry vault places spent nuclear fuel in vertically oriented cylindrical steel fuel storage containers which are then inserted into a steel charge face structure within the thick concrete structure. A labyrinth airflow passage system provides natural convection airflow for decay heat removal. The shielding is provided by the 106.7 cm (42 in) thick concrete walls and the labyrinth airflow passages. For the Fort St. Vrain fuel, maximum design modular dry vault surface dose rate is 21 mrem/hr. A picture of the cross section of the modular dry vault is shown in Figure F-6.

NRC Certification or Basis for License

The modular dry vault model has been approved by the NRC for the site-specific application at Fort St. Vrain. The basis for the license is 10 CFR 72.

F.1.1.2.4.7 Ventilated Storage Cask System (VSC-24)

Description of VSC-24

The Ventilated Storage Cask, designed by Sierra Nuclear Corporation, is a vertical concrete cask design that has been approved by the NRC and is in use at the Palisades nuclear power plant site. As with the NUHOMS design, this system relies on concrete for both neutron and gamma shielding and incorporates internal airflow passages requiring detailed shielding analyses to demonstrate acceptable streaming doses. The Ventilated Storage Cask design dose rates are 20 mrem/hr side contact and 50 mrem/hr top contact. A sketch of the Ventilated Storage Cask is shown in Figure F-7.

NRC Certification or Basis for License

The Sierra Nuclear Corporation's Model VSC-24 has been granted Certificate of Compliance Number 1004 by the NRC. The basis for this certificate is 10 CFR 72 Subparts L and K.

F.1.1.2.5 Manufacturers of Commercial Nuclear Fuel Dry Storage Systems Not Currently Licensed by the NRC in the United States

In addition to the above examples of dry cask storage systems licensed in the United States, there are other systems either licensed outside the United States or in the design and/or licensing stage (Table F-6). Tables F-7 and F-8 show shielding and thermal related parameters of the various dry cask models that are not currently licensed in the United States.

F.1.1.2.5.1 Description of MACSTOR

The MACSTOR system (designed by Atomic Energy of Canada, Ltd. and Transnuclear, Inc.), representing a synthesis of both metal and concrete casks in a modular dry vault, is being reviewed for use in Canada (AECLT, 1994). Spent nuclear fuel is placed in 0.95 cm (0.375 in) thick carbon steel canisters or baskets that are then placed (in a vertical position) in concrete modules. Air labyrinth passages into and

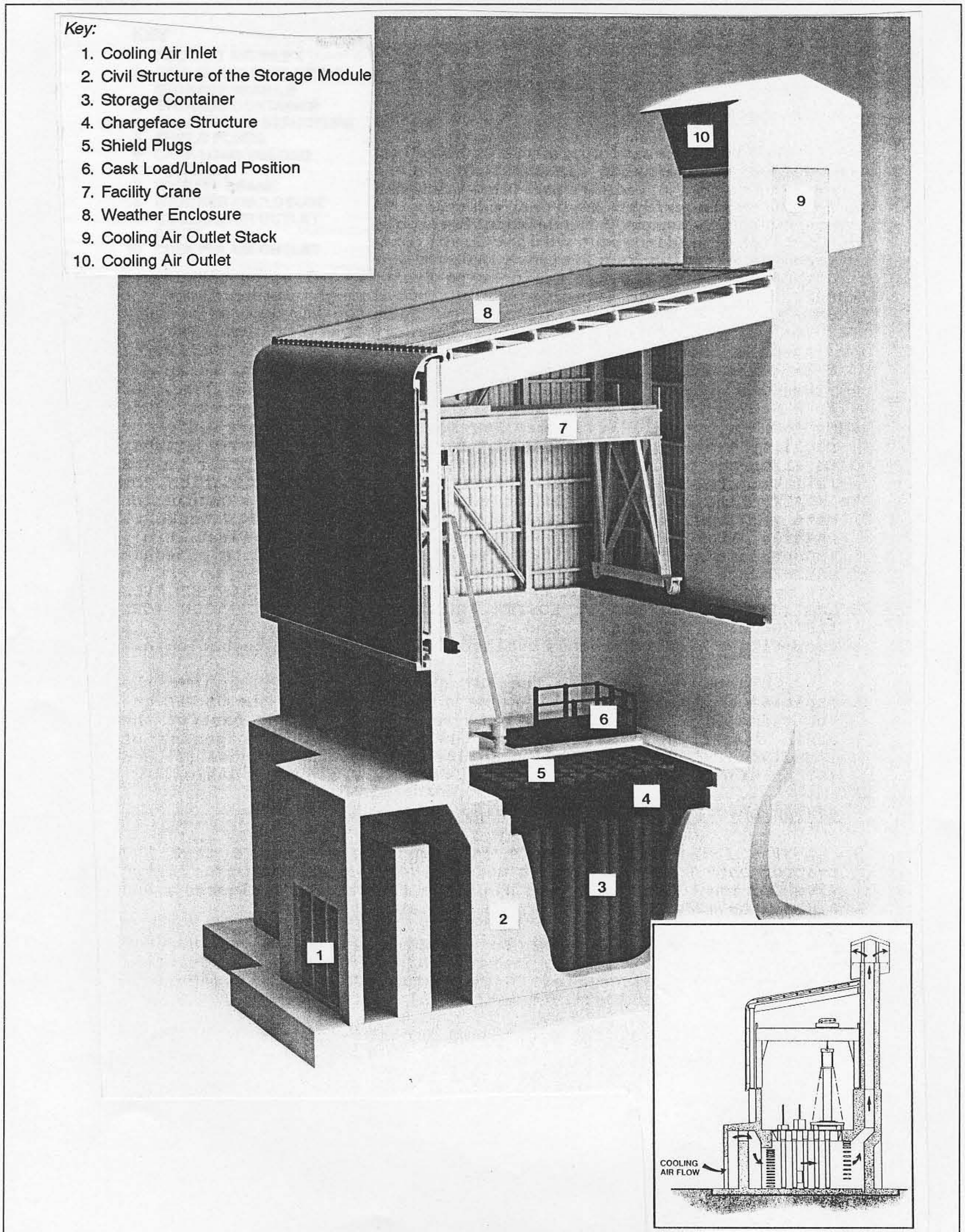


Figure F-6 Photograph of a Single Modular Dry Vault Module

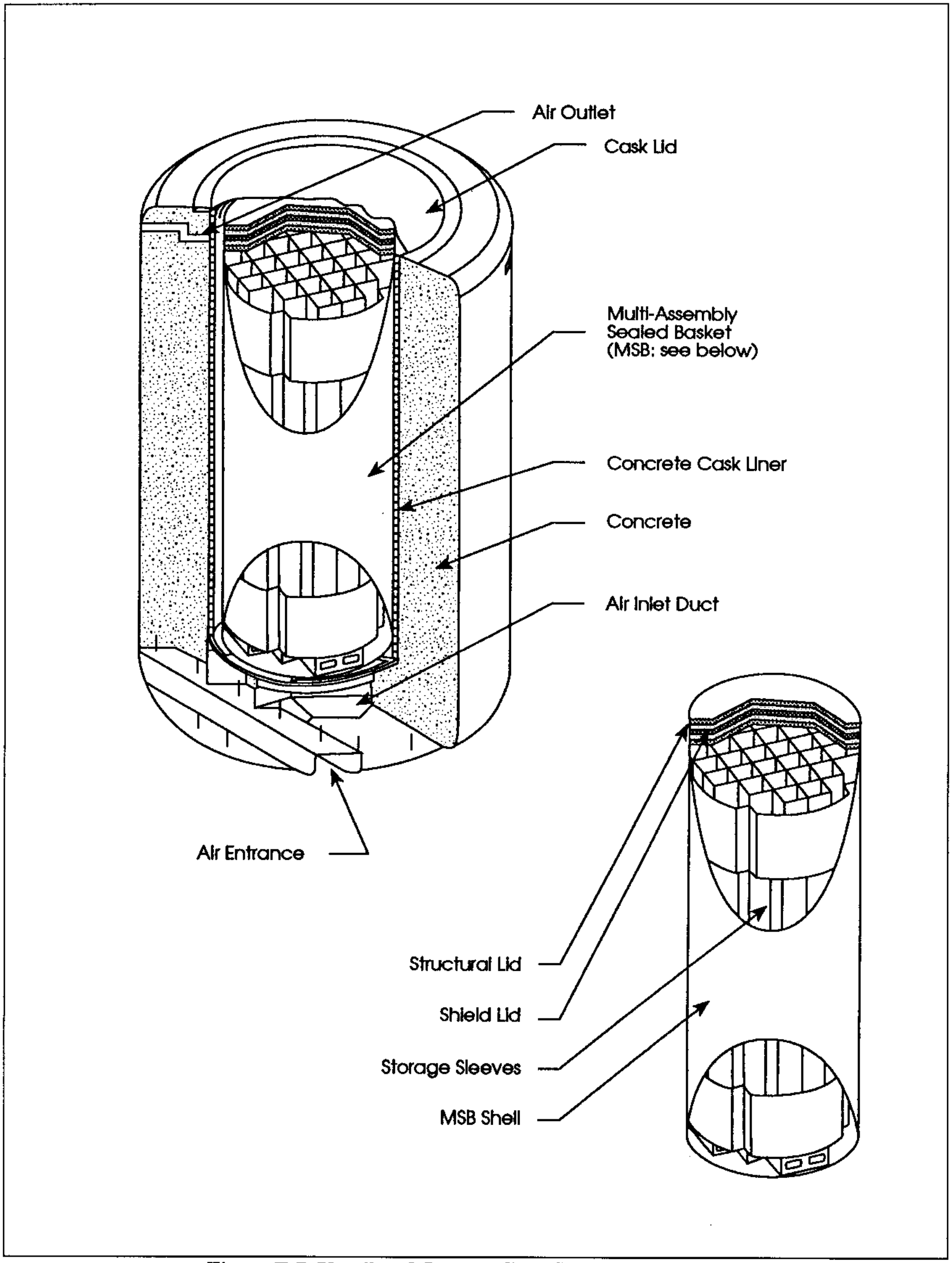


Figure F-7 Ventilated Storage Cask System Components

Table F-6 Manufacturers of Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Facility or Model*	*Location or Status*
General Nuclear Systems, Inc. of Ontario Hydro	HDC	Canada
Atomic Energy of Canada, Ltd./Transnuclear Inc.	MACSTOR	Canada
Siemens Power Corporation	FUELSTOR	Germany
RISO National Laboratory	DR3	Denmark
Atomic Energy of Canada, Ltd.	SILO, Canister	Canada
Nuclear Assurance Corporation	NAC-26	10 CFR 72 License Pending
General Nuclear Systems, Inc.	X28 and X33	10 CFR 72 License Pending
Transnuclear, Inc.	TN-32	10 CFR 72 License Pending

FUELSTOR = Fuel Encapsulation and Lag Storage

Table F-7 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Model*	*Shield Material*	*Shield Thickness in cm (in)*	*Design Limit Surface Dose Rate milliSievert/hr (mrem/hr)*
GNSI/OH	HDC-36	High Density Carbon Steel, Concrete	2.5 (1) CS, 45.7 (18) High-Density Concrete	NA
Siemens Power Corporation	FUELSTOR	Concrete	304.8 (120)	0.00001 (0.001)
Transnuclear/Atomic Energy of Canada Ltd.	MACSTOR	Concrete	96.5 (38)	0.025 (2.5)
RISO National Laboratory	DR3	Carbon Steel, Concrete, Earth	76.2 (30) Carbon Steel, Axial 15.2 (6) Concrete, Radial	NA
Atomic Energy of Canada Ltd.	SILO	Concrete	91.4 (36) Concrete	0.025 (2.5)
Atomic Energy of Canada Ltd.	CANSTOR	Concrete	NA	NA
Transnuclear	TN-32	Borated Resin Carbon Steel	Radial: 21.6 (8.5) 11.4 (4.5) Resin Bottom: 22.2 (8.75) Carbon Steel Top: 15.9 (6.25) Cast Iron	Side: 0.86 (86) Top: 0.18 (18) Bottom: 3.15 (315)
General Nuclear Systems, Inc.	CASTOR X28 and X33	Cast Iron, Stainless Steel Polyethylene Rods	Radial: 30.5 (12) Bottom: 27.9 (11) Top: 39.1 (15.4) Rods Radial: 72- 6.1 (2.4) Dia.	2 (200)

NA = Not Available; FUELSTOR = Fuel Encapsulation and Lag Storage ; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro; Transnuclear = Transnuclear, Inc.

out of the module provide a flow path for natural convection airflow to remove decay heat. The vault concrete walls are 96.5 cm (38 in) thick and designed to reduce dose rates to less than 2.5 mrem/hr on contact. This concrete thickness is maintained even where the airflow passage labyrinth is located. A cross section of the MACSTOR is given in Figure F-8.

F.1.1.2.5.2 Description of a Fuel Encapsulation and Lag Storage Facility

The Fuel Encapsulation and Lag Storage system is similar to the modular dry vault and MACSTOR/CANSTOR in that it is a stand-alone concrete building with interior steel storage containers. Unlike the modular dry vault and CANSTOR/MACSTOR, the Fuel Encapsulation and Lag Storage system

Table F-8 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	*Model*	*Design Heat Load (KW)*	*Thermal Limits (°C)*	*Heat Transfer Mode(s)*
GNSI/OH	HDC	---	---	Conduction, Radiation
Siemens Power Corporation	FUELSTOR	Up to 2 KW Per Canister	380/2 KW 250-1KW-Clad	Conduction, Radiation, Natural Convection
Transnuclear Inc./Atomic Energy of Canada Ltd.	MACSTOR CANSTOR	240 (20 Canisters at 12 each)	93 Concrete 340 LWR Clad	Conduction, Radiation
RISO National Laboratory	DR3	---	NA	Conduction, Radiation, Forced Convection
Atomic Energy of Canada Ltd.	SILO	4	NA	Convection
Transnuclear Inc.	TN-32	---	NA	Conduction, Radiation
General Nuclear Systems, Inc.	CASTOR X28 and X33	19.2 and 20.9	370 LWR Cladding	Conduction, Radiation

NA = Not Available; LWR = Light Water Reactor; FUELSTOR = Fuel Encapsulation and Lag Storage; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro

stores spent nuclear fuel containers in a horizontal position. The Fuel Encapsulation and Lag Storage system is designed for a surface contact dose rate of 0.001 mrem/hr, and relies on its combined 304.8 cm (120 in) thick inner and outer shield concrete walls and labyrinth airflow passages for shielding. The Fuel Encapsulation and Lag Storage system is not licensed by the NRC or in use in the United States A cross section of the Fuel Encapsulation and Lag Storage system is shown in Figure F-9.

F.1.1.2.5.3 Description of a RISO National Laboratory Facility

In Denmark, the RISO National Laboratory has designed and constructed a dry storage facility at its DR3 PLUTO type research reactor to store MTR spent nuclear fuel from this reactor. This facility was installed under the floor of the active handling bay at the reactor and consists of four prefabricated octagonal concrete blocks placed in a vertical position into steel lined holes in the earth. Each block contains 12 storage holes in a triangular mesh, with a carbon steel form forming each hole and a separate stainless steel tube containing the spent nuclear fuel. Axial shielding is provided by a 76.2 cm (30 in) thick carbon steel plug. Radial shielding is provided by the surrounding earth and concrete of the octagonal block with the minimum concrete thickness of 15.2 cm (6 in). A sketch of the RISO National Laboratory's design is shown in Figure F-10.

F.1.1.2.5.4 Description of SILO

The SILO has been designed and licensed in Canada, and over 180 concrete SILOs have been built for the storage of Canadian research reactor and CANDU-commercial reactor spent fuel (AECLT, 1994). The SILO consists of a carbon steel-lined cylindrical hole inside a 91.4 cm (36 in) thick vertical concrete cylinder without any labyrinth airflow passage for heat removal. Carbon or stainless steel canisters containing the spent nuclear fuel are placed inside the SILO and stacked up to nine high before being covered by a steel and concrete plug. The surface SILO dose rate limit is 2.5 mrem/hr. The unique design aspect of the SILO is that it is the only concrete cask without airflow passages for natural convection heat removal. It has been used for short-length low decay heat fuel which is dimensionally similar to foreign

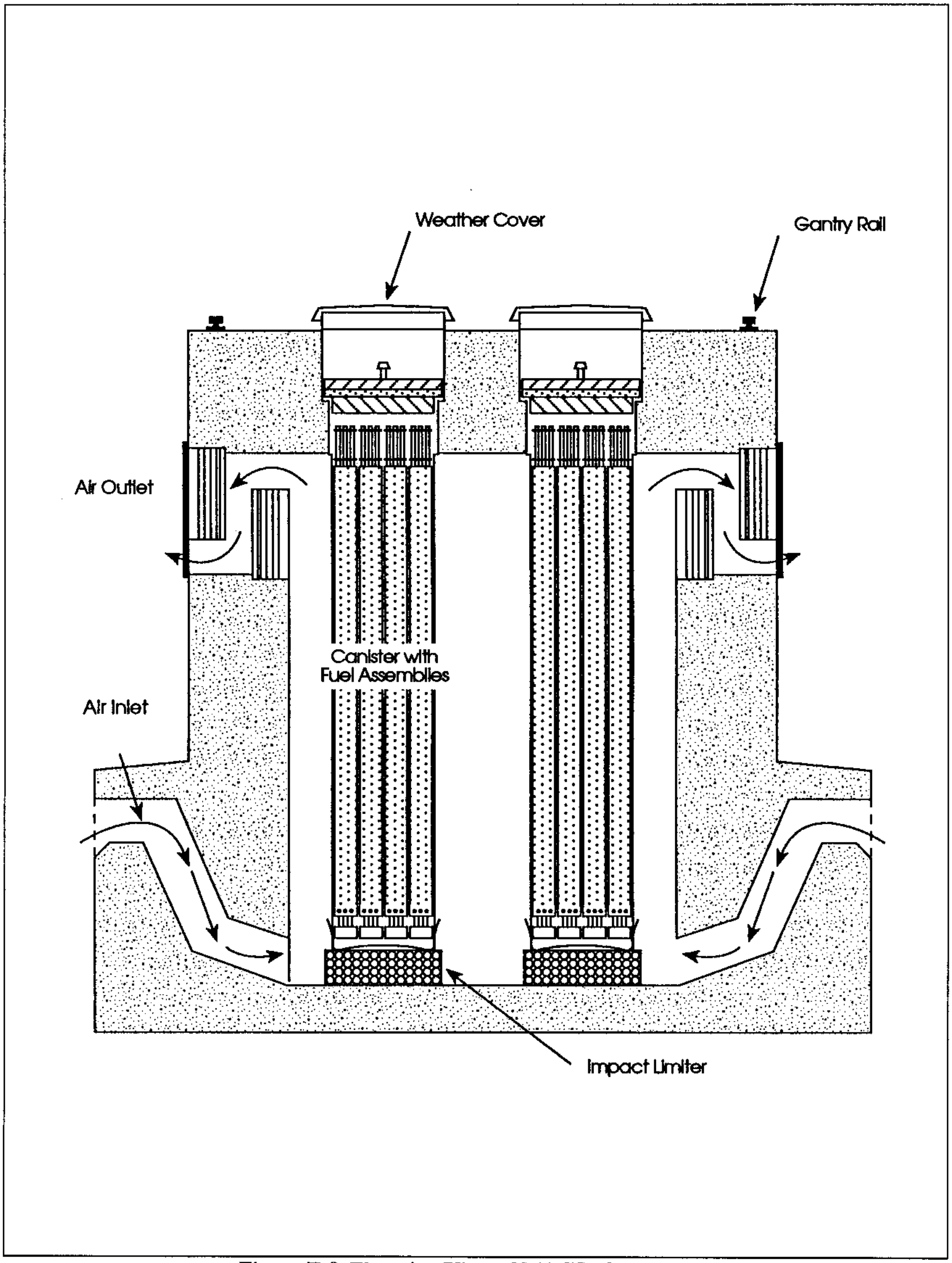


Figure F-8 Elevation View of MACSTOR Module

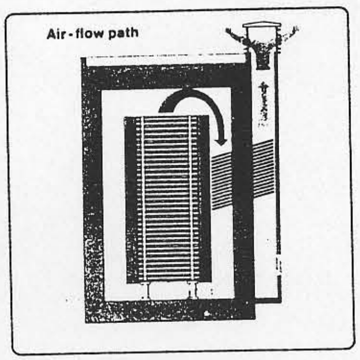
FUELSTOR

FUEL ENCAPSULATION AND LAG STORAGE FACILITY

For at-reactor dry storage of spent fuel

Main technical features:

- Closed-cycle vault resulting in triple barrier system
- Passive cooling by natural convection ensuring low fuel temperatures
- Low initial storage temperature resulting in no restrictions regarding the length of the storage period
- Sealed canister storage
- Horizontal placement of helium-filled canisters in storage vault
- Reinforced concrete construction for safe shielding resulting in radiation levels outside the building below permissible dose of unrestricted area (ALARA)
- Reinforced concrete construction to withstand the effects of natural phenomena and man-induced events such as earthquakes, tornado missiles and aircraft crashes
- Nuclear criticality safety is ensured by the building design as well as spent fuel canister arrangement
- Store is easily expandable by modules
- Low land consumption due to compact spent fuel storage
- Nuclear safeguard provisions



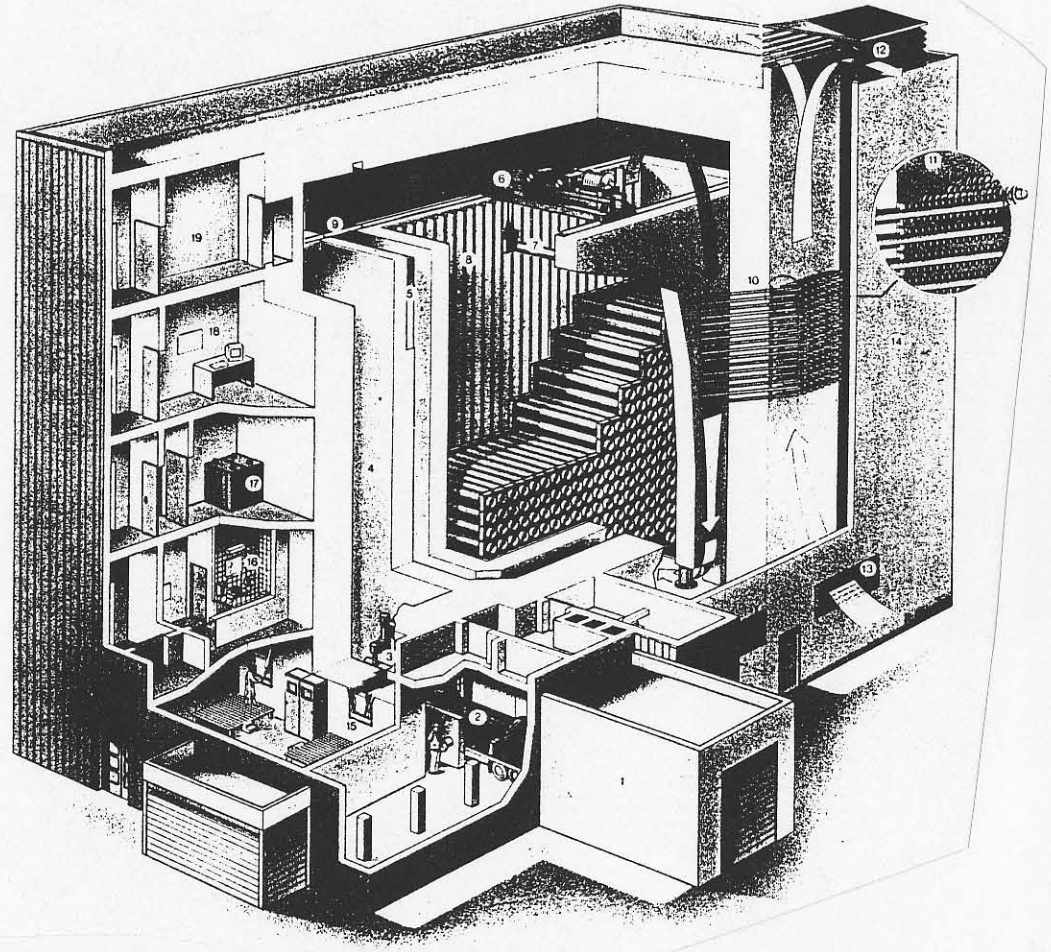


Figure F-9 Fuel Encapsulation and Lag Storage System Cross Section

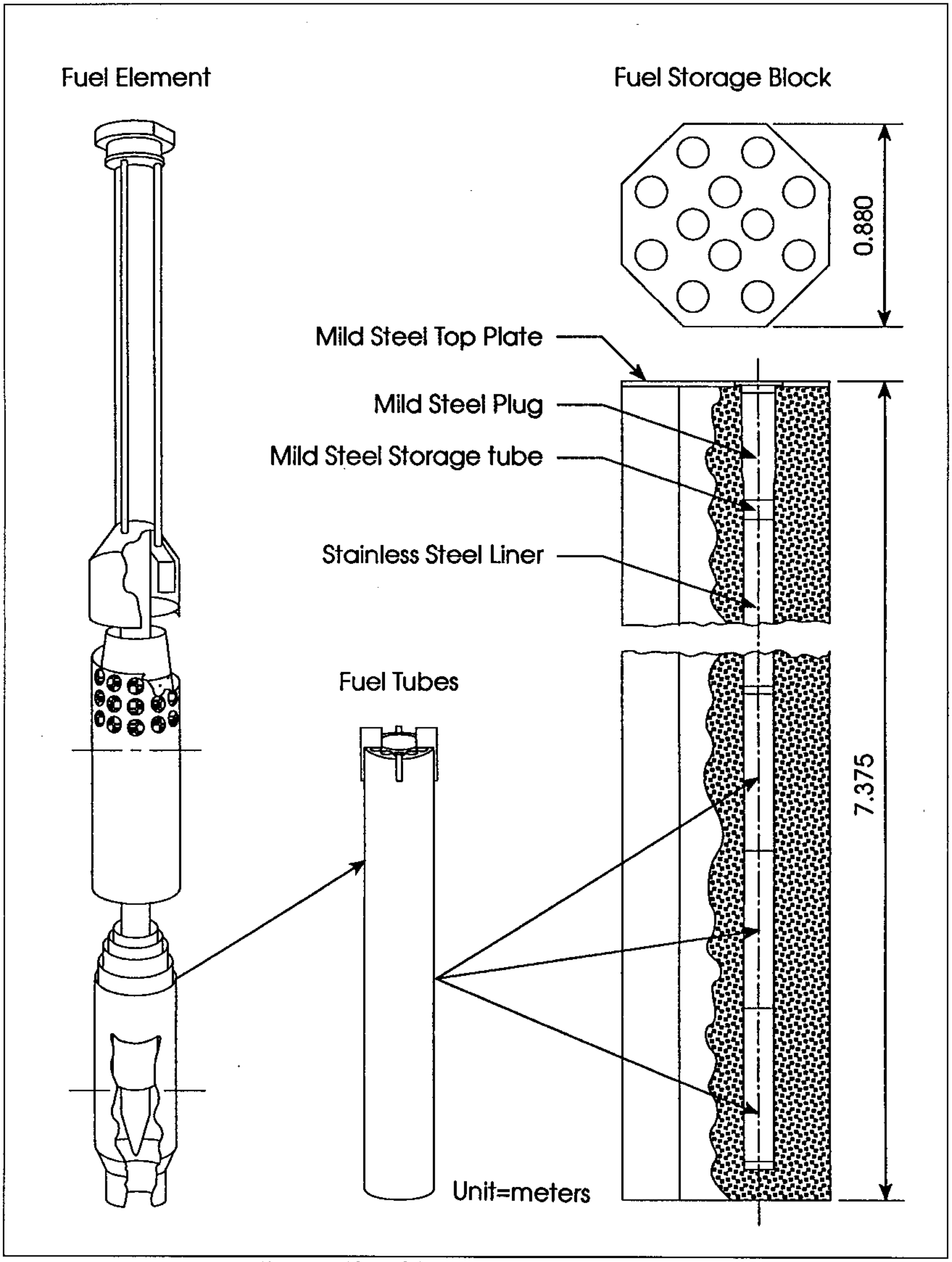


Figure F-10 RISO National Laboratory Design

research reactor spent nuclear fuel. The SILO has been licensed in Canada and is currently undergoing license approval in South Korea, which uses the same regulations as the NRC. A sketch of the SILO is given in Figure F-11.

F.1.1.2.5.5 Dual-Purpose Cask and Canister Systems

Dual-purpose designs must satisfy NRC requirements for both storage and transportation (10 CFR Parts 72 and 71, respectively). It is believed that such dual-purpose designs would reduce incident-free handling of individual spent nuclear fuel assemblies, reduce the volume of low-level radioactive waste that would otherwise be generated from using a single-purpose cask system (one cask for storage with subsequent transfer of individual assemblies to transport casks and disposal packages), and may play a role in reducing overall worker radiation exposures over a single-purpose cask system.

At the present time, there are two dual-purpose casks for light water reactor fuel use: Nuclear Assurance Corporation's Storage/Transport Cask and Vectra's dual-purpose canister system (MP-187). Nuclear Assurance Corporation's Storage/Transport Cask has received NRC approval. The NRC is expected to approve Vectra's MP-187 in the near future.

The VECTRA MP-187 is a derivative of a design approved earlier by the NRC. The MP-187 design includes a stainless steel confinement canister, a horizontal reinforced concrete module for storing the canister, and a special onsite/offsite transportation cask system that may also be used to store the canister in a vertical orientation. This system is currently being evaluated by the NRC for the Rancho Seco nuclear power plant. The applicant also has a variation for the canister design to accommodate canned spent nuclear fuel for damaged spent nuclear fuel assemblies, and which cannot be stored without a second confinement barrier.

DOE had proposed expanding the role of a dual-purpose system to that of a multi-purpose canister-based system (DOE, 1994f; DOE, 1994b; DOE, 1994c). Fuel would be loaded into a canister at the reactor site. The canister could then be placed into unique, specially designed overpacks for storage at the reactor site, transportation to a federal facility, or disposal in a repository. Final NRC approval for use of the multi-purpose canister as a component of the disposal package requires that the multi-purpose canister and its surrounding overpack meet 10 CFR Part 60 requirements. The fact that no site has been chosen yet for a repository adds an element of uncertainty to the third function: disposal. DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this canister. The Department of the Navy, however, will complete this EIS and will limit its scope to the storage and transport of Navy spent nuclear fuel.

F.1.2 Wet Storage Designs

In addition to the previous examples of dry storage technology, there are several types of wet storage systems currently in use at DOE sites and at commercial nuclear power facilities. These include aboveground pools (lined or unlined), inground pools (lined or unlined), and shutdown reactor vessels. For the purposes of this appendix, a pool refers to a canal or a basin.

Description of Wet Pool Spent Nuclear Fuel Storage Technology

The storage of spent nuclear fuel in pools (i.e., wet storage technology) has been in use for over 40 years (since the early water-cooled reactors began operating). The basic concept underlying wet storage is analogous to the development of light water-cooled nuclear reactors for defense, research, and electric power production purposes.

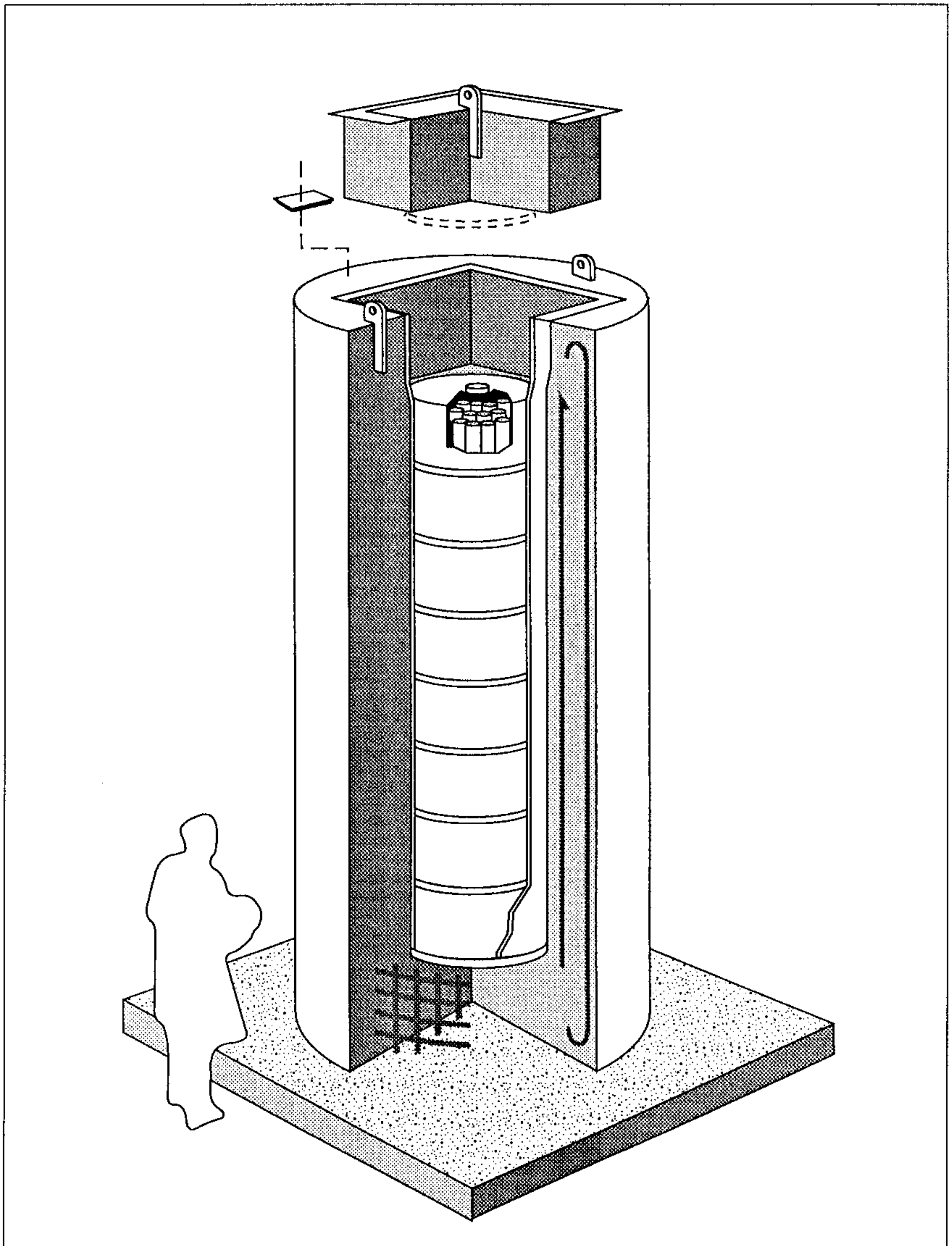


Figure F-11 The Atomic Energy of Canada, Ltd. Concrete SILO

In terms of spent nuclear fuel storage, water offers several distinct advantages, which can be summarized as:

- low cost for shielding and coolant medium,
- visual confirmation of fuel location and ease of handling,
- high heat capacity allowing for a large time period before thermal limits are exceeded,
- multi-purpose shield for both neutrons and gamma rays,
- inherent ability to retain many fission products which could leak from failed spent nuclear fuel, and
- insusceptibility to degradation from spent nuclear fuel radiation.

Water pool storage also has some shortcomings. These are:

- the need to maintain high purity water to prevent corrosion,
- the requirement for active safety systems connected to the water for heat removal, purity control, and water makeup,
- extensive lined and reinforced concrete walls for ensuring no leakage of water under all accident conditions,
- generation of radioactive waste from degraded fuel which is collected by the water purification systems, and
- groundwater monitoring to detect any leakage of radioactive pool water into the environment.

For every water-cooled reactor in the world, the decision has always been made to construct an adjoining or integral spent nuclear fuel storage pool. Currently, over 600 water cooled electric power-, research-, and defense-related reactors are operating in the world, each with its own wet storage pool for spent nuclear fuel (Nuclear Engineering International, 1993). Experience has shown that this technology is safe and effective.

At commercial nuclear power plants, the pool storage is located in a structure adjacent to a containment building that is capable of direct hydraulic connection to the reactor core through a system of canals, gates, and pools. The spent nuclear fuel pool building is designed and built to withstand all the accidents and dynamic loads required of other safety-related structures at nuclear power plants. It has its own crane and fuel handling equipment, and a separate heating, ventilation, and air conditioning system to mitigate radioactive releases to the environment. The nuclear power plant control room includes monitors and controls for the spent nuclear fuel pool. Redundant separate trains of equipment are used to fulfill the requirements of heat removal from the spent nuclear fuel pool water, removal of impurities and radioactive materials from the water, and maintenance of the water level to ensure adequate shielding above the spent nuclear fuel. At commercial power plants, such parameters as water level, water temperature, flow and temperature difference across heat exchangers used to cool the water, water purity, activity levels, and radiation dose rates are all monitored and measured.

All U.S. commercial nuclear power plant pools are stainless steel lined and use racks made of stainless steel to store spent nuclear fuel. Stainless steel is used to line the pool walls and floor to help maintain high water purity by preventing the release of chemicals from unlined concrete and to simplify decontamination at the end of the facility's life. The racks provide support and spacing for each fuel assembly, thus controlling criticality and maintaining fuel structural integrity.

Detailed criticality and thermal-hydraulic analyses are performed to demonstrate to the licensing authorities (the NRC in the United States) that fuel can never become critical, and that the assembly spacing in the racks allows for adequate cooling so as to prevent nucleate and bulk boiling in the pool or on any fuel surfaces. Shielding analyses substantiate the adequacy of the water depth above the fuel in the pool (usually at least 6.1 m or 20 ft), and the thickness of concrete pool walls and piping routing for systems connected to the pool water. This piping may contain pool water that is contaminated with radioisotopes released from spent nuclear fuel in the pool, and must be considered in dose rate evaluations. The shielding analyses provide assurances that the dose rate levels are acceptably low to workers around the spent nuclear fuel pool. Accident analyses are performed to show that the most conservative effects to the public of a postulated release of failed spent nuclear fuel fission products in the pool meet all regulatory dose rate limits.

One difference between nuclear power plant spent nuclear fuel wet storage and that which would be used for foreign research reactor spent nuclear fuel is that at nuclear plants, the pools include soluble boron in the water as a means of controlling criticality. Boron is a powerful neutron absorber, and as such prevents the approach to criticality since neutrons are needed to initiate and maintain a uranium fission chain reaction. Soluble boron would not be used in a wet storage facility for foreign research reactor spent nuclear fuel because it would corrode the aluminum cladding materials present in most foreign research reactor spent nuclear fuel. If neutron-absorbing materials were deemed desirable for foreign research reactor spent nuclear fuel wet storage, they could be incorporated as solid boron-aluminum alloy plates encased in aluminum or stainless steel that are integral to the storage rack design so that boron is physically present between each fuel assembly. This design has been successfully licensed and operated at many commercial nuclear power plants to allow for a higher density or tighter packing of the spent nuclear fuel assemblies.

Figure F-12 illustrates a typical wet pool storage facility for spent nuclear fuel. The operational experience of wet storage facilities is excellent, with no significant accidents or events. In the few instances when fuel was damaged while being moved into or out of the pool, the water mitigated any radiological consequences to workers, the public, or the environment. Some events involved temporary loss of pool water heat-removal systems. The large heat capacity of the water in the pool reduced any increase in fuel temperature so that no harmful effects resulted from such a loss in cooling capacity. These two types of wet storage facility events emphasize the principal benefit of water as a coolant and shielding medium, namely its very large thermal inertia and shielding/radionuclide retention capacity.

There is a long and successful history of safe operation for wet storage of spent nuclear fuel in the commercial power, research, and defense sectors of the nuclear industry. The technology is well known, licensed, and offers extensive operational experience.

The same arguments that apply to dry storage of spent nuclear fuel also apply to wet storage facilities. 10 CFR 72 applies to both dry and wet storage facilities for spent nuclear fuel at commercial licensees. DOE Order 6430.1A (DOE, 1989a) applies to the general design of nuclear facilities on DOE sites, including those for spent nuclear fuel storage. This DOE order references 10 CFR 72 for most of the specific details on spent nuclear fuel storage.

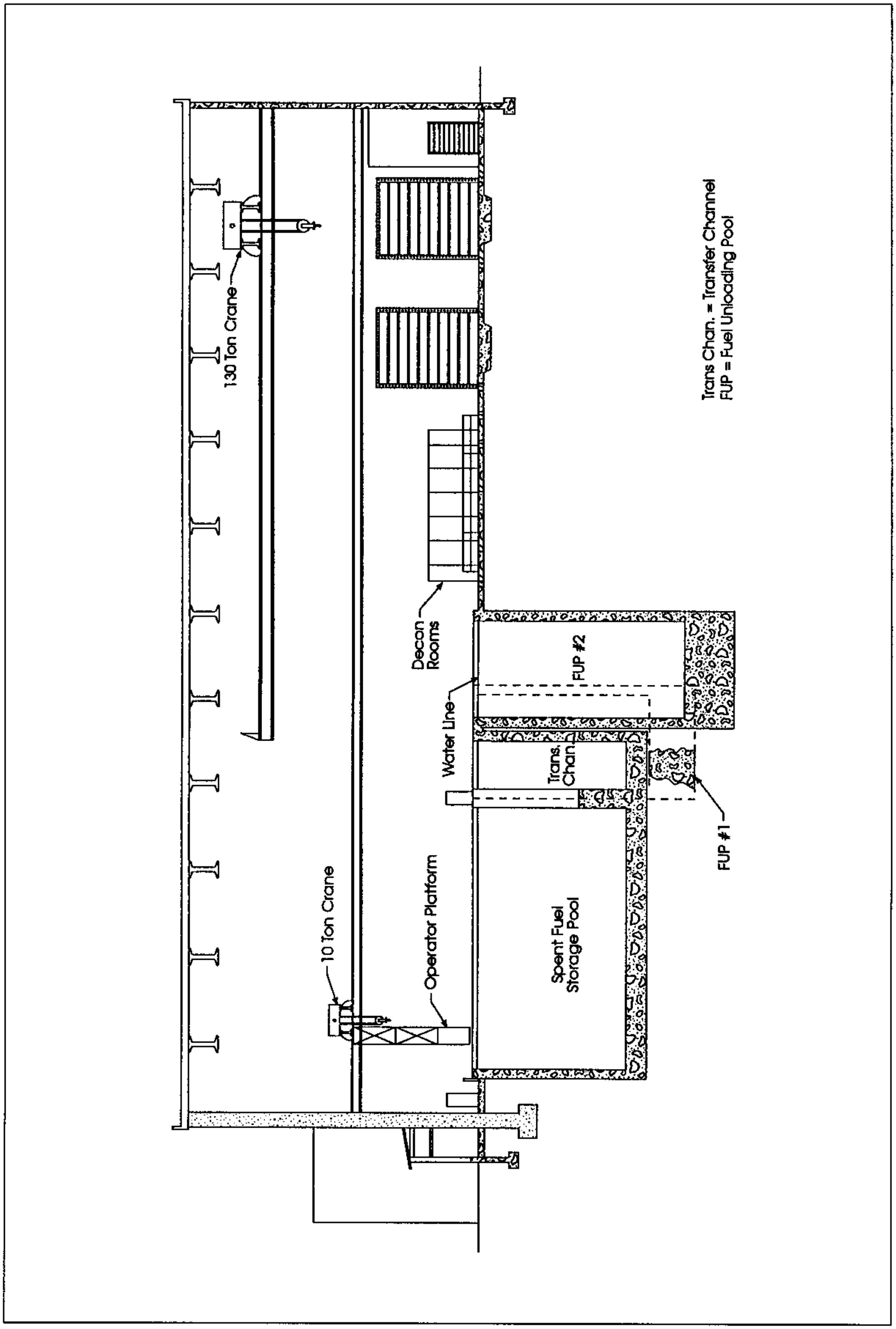


Figure F-12 Typical Wet Pool Storage Facility for Spent Nuclear Fuel

F.1.3 Summary of DOE Spent Nuclear Fuel Locations and Activities

DOE currently has about 2,700 MTHM of spent nuclear fuel in its storage facilities across the DOE complex (DOE, 1994h). Additional generation of about only 100 MTHM is anticipated during the next 40 years. Most of the spent nuclear fuel storage occurs at three sites: Hanford Site (77 percent), Idaho National Engineering Laboratory (10.9 percent), and Savannah River Site (7.3 percent) (Table F-9). Note that the quantities of DOE spent nuclear fuel completely dwarf the expected amount of foreign research reactor spent nuclear fuel (about 19 MTHM) on an MTHM basis (i.e., foreign research reactor spent nuclear fuel is less than 1 percent of the total). However, on a volume basis, foreign research reactor spent nuclear fuel represents about 10 percent of the total and, thus, their storage facilities would be of a significant size. Predominantly wet storage is used at DOE sites, although some limited experience exists with dry storage (e.g., Los Alamos National Laboratory and Idaho National Engineering Laboratory).

Table F-9 DOE Spent Nuclear Fuel Inventory[a, b]

Generator or Storage Site[c]	*Existing (1995)*		*Future Increases (through 2035)*		*Total (2035)*	
	MTHM[d]	*Percent*	*MTHM[d]*	*Percent*	*MTHM[d]*	*Percent*
Hanford Site	2,132.44	80.6	0.00	0.0	2,132.44	77.8
Idaho National Engineering Laboratory[e]	261.23	9.9	12.92	13.5	274.14	10.0
Savannah River Site	206.27	7.8	0.00	0.0	206.27	7.5
Naval Nuclear Propulsion Reactors	0.00[f]	0.0	55.00	57.6	55.0	2.0
Oak Ridge Reservation	0.65	<0.1	1.13	1.2	1.78	<0.1
Other DOE Sites	0.78	<0.1	1.50	1.6	2.28	<0.1
Non-DOE Domestic Research Reactors[g]	2.22	<0.1	3.28	3.4	5.50	0.2
Special-Case Commercial Reactors[h]	42.69	1.6	0	0	42.69	1.6
Foreign Research Reactors[i]	0	0	21.7	22.7	21.70	0.8
Total	**2,646.27**		**95.53**		**2,741.80**	
Percent of 2035 Total	**96.5**		**3.5**		**100.00**	

[a] *Source: DOE, 1995g*

[b] *Numbers may not sum due to rounding.*

[c] *The Nevada Test Site does not currently store or generate spent nuclear fuel and is not expected to generate spent nuclear fuel through 2035. However, in the 2010-2020 timeframe, a repository may open, with annual capacity over 1,000 MTHM.*

[d] *One MTHM equals approximately 2,200 pounds.*

[e] *Sum of fuel located at the Idaho National Engineering Laboratory.*

[f] *Existing inventory of Naval spent nuclear fuel is included in the Idaho National Engineering Laboratory totals (9.95 MTHM).*

[g] *Includes research reactors at commercial, university, and Government facilities.*

[h] *This total is just that stored at non-DOE facilities (Babcock & Wilcox Research Center and Fort St. Vrain). The total inventory of spent nuclear fuel from special-case commercial reactors is 186.41 MTHM. This fuel is also stored at the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, the Savannah River Site, and the West Valley Demonstration Project.*

[i] *At the Savannah River Site and the Idaho National Engineering Laboratory.*

Wet Storage

DOE spent nuclear fuel pools are in many cases more than 20 years old and were originally unlined, due to simplicity and the relatively short planned duration (3 to 6 months) of spent nuclear fuel storage prior to reprocessing. The spent nuclear fuel storage basins are concrete with 30 to 90 cm (1 to 3 ft) thick walls, and the bottom is usually thicker than the sides. For shielding purposes, the pool maintains a minimum of 3 m (10 ft) of water over the spent nuclear fuel at all times. Thus, total water depth typically ranges between 4.5 to 6.1 m (15 to 20 ft), although some facilities extend to 9.1 m (30 ft). Steel, stainless steel, or aluminum racks are affixed to the bottom of the pool for holding the spent nuclear fuel in a vertical configuration. The spent nuclear fuel basins provide for recirculation and heat removal capabilities, but limited water clarification and purification. Chemicals from the exposed concrete increase pool turbidity and tend to accelerate corrosion phenomena, particularly for aluminum-clad fuels. Some of the DOE spent nuclear fuel wet storage facilities do not meet the present construction requirements.

Wet storage still remains the predominant technology for storing irradiated materials (DOE, 1993b; Taylor et al., 1994). Currently, there are some 29 DOE spent nuclear fuel storage pool facilities in the complex, ranging in age from 10 years to more than 40 years. Facilities built more than 30 years ago were constructed to standards far less rigorous than exist today. Several DOE orders address spent nuclear fuel storage facilities indirectly, while DOE Order 6430.1A specifically sets the design criteria that addresses storage facilities (spent nuclear fuel facilities that are part of a reactor facility are covered by DOE Order 5480.6). Most DOE storage pools were not designed for long-term storage of spent nuclear fuel and targets and have very limited space available for consolidation.

Most of the storage pool surfaces are bare concrete A few are lined with stainless steel, and some are coated with epoxy or vinyl. The unlined pools are more susceptible to leakage and to increased contamination by soluble radionuclides. The unlined bare concrete storage pools do not have effective leak-detection systems to detect and capture potential leaks. To help identify pool leakage, more than 50 percent of DOE storage pools have had groundwater monitoring wells installed.

Severe corrosion of materials within many of the DOE storage pools has occurred. Corrosion has been generally attributed to poor water quality control and material incompatibilities, which has led to pitting and galvanic corrosion of spent nuclear fuel and storage equipment. This could potentially create a problem when the spent nuclear fuel materials have to be moved. In some cases, equipment failure could cause fissile material reconfiguration, which could increase nuclear criticality concerns. As a result of corrosion, release of radionuclides and fissile material to the pools has occurred. Corrosion also creates handling, packaging, inventory control, waste generation, and cleanup problems with the storage pools.

Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities potentially suitable for near-term future wet spent nuclear fuel storage (in some cases with facility upgrades):

- Idaho National Engineering Laboratory
  - Power Burst Facility Canal
  - Idaho Chemical Processing Plant (ICPP)-666 Pool
  - Expended Core Facility
- Savannah River Site
  - 105-K Disassembly Basin

- 105-L Disassembly Basin
- 105-C Disassembly Basin
- 105-P Disassembly Basin
- Receiving Basin for Offsite Fuels (RBOF) Facility (244-H)
- BNFP (acquisition required).

However, only the ICPP-666 pool and the BNFP were found to meet all current standards, and, thus, be considered suitable for long-term storage.

DOE has improved some of its spent nuclear fuel facilities and has plans for additional upgrades (DOE, 1993b; DOE, 1995g). Typical upgrades include:

- installation and operation of water purification equipment, such as demineralizer columns and filters,
- reracking and fuel consolidation to increase fuel storage space, and
- improving seismic resistance (where possible, via additional supports).

These upgrades would extend the life of existing facilities and allow safe storage of spent nuclear fuel until new facilities are constructed or the spent nuclear fuel is chemically separated. In addition, spent nuclear fuel suspect of leaking during this interim period would be removed and canned to extend its safe storage.

Dry Storage

DOE has fewer dry storage facilities, and these range from approximately 1 to 50 years in age. There are many different types and applications of dry storage used throughout the DOE complex. Spent nuclear fuel is sorted in steel structures; lined and unlined concrete hot cells; steel-lined; concrete; below-grade vaults; reprocessing canyon dissolver cells; cans contained in steel wells; and large, above-grade storage casks. Spent nuclear fuel has been characterized and stored in dry configurations within hot cell facilities since the 1950s. Most DOE hot cells were not designed and built for long-term storage of spent nuclear fuel. Their primary mission was to conduct tests and basic research on irradiated fuels resulting in very limited capacity for storage of spent nuclear fuel.

Since the 1970s, spent nuclear fuel has been stored in facilities specifically engineered for longer-term dry storage. Modern dry storage methods in newer facilities provide low corrosion environments within sealed barriers for monitored interim retrievable storage. A few examples of dry storage confinement methods include sealed canisters in wells surrounded by concrete and extensive release protection incorporating High Efficiency Particulate Air-filtered ventilation systems. By using current dry storage technology, dry storage facilities could be engineered to withstand severe natural phenomena hazards, fires, and explosions without damage to the fuel or release of radionuclides. Dry storage technologies can be adapted to store many types of damaged and undamaged DOE-owned spent nuclear fuel.

The application of dry storage technologies generally results in fewer environmental, safety, and health issues as compared with wet storage. However, DOE has limited experience with aluminum-clad, high decay heat fuels in dry storage facilities.

Some quantities of spent nuclear fuel may be in dry storage facilities for much longer than originally anticipated. Over the years, several inground steel-lined storage well barriers have had the potential for severe corrosion, which could result in undetected releases to the environment. This is particularly important, because of the inaccessibility of these facilities for inspection and characterization (e.g., Argonne National Laboratory Radioactive Scrap and Waste Facility).

The Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities suitable for near-term dry storage of spent nuclear fuel.[1]

- Argonne National Laboratory-West
  - Hot Fuel Examination Facility
  - Radioactive Scrap and Waste Facility
- Idaho National Engineering Laboratory
  - Test Area North Test Pad
  - ICPP Irradiated Fuel Storage Facility (IFSF)
  - ICPP-749 (Drywells, Second Generation)
- Savannah River Site
  - 221-H (Extensive modification required)
  - 221-F (Extensive modification required).

However, certain DOE requirements, such as DOE Orders 6430.1A and 5480.6, make it likely that these facilities could not qualify for future, long-term, dry storage of spent nuclear fuel. Excluding the Savannah River Site facilities (because of the extensive required modifications), none of these facilities appear to be very useful for long-term spent nuclear fuel storage. Extensive modifications to the facilities would be required to meet seismic criteria and increase the storage capacity or convert existing facilities (e.g., F- and H-Canyons at the Savannah River Site) into suitable dry storage facilities. However, facilities such as the Hot Fuel Examination Facility and Test Area North appear suitable as possible staging and characterization facilities in a dry cask storage approach, based upon the presented information (Taylor et al., 1994).

F.1.3.1 Savannah River Site

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, Georgia (DOE, 1995g). The Savannah River forms the southwestern border of the Savannah River Site. The Savannah River Site consists primarily of managed upland forest with some wetland areas, and facilities and railways occupy approximately five percent of the Savannah River Site land area. Figure F-13 presents a map of the Savannah River Site with spent nuclear fuel facilities displayed.

1 *Existing facilities in Nevada were not included in the analysis.*

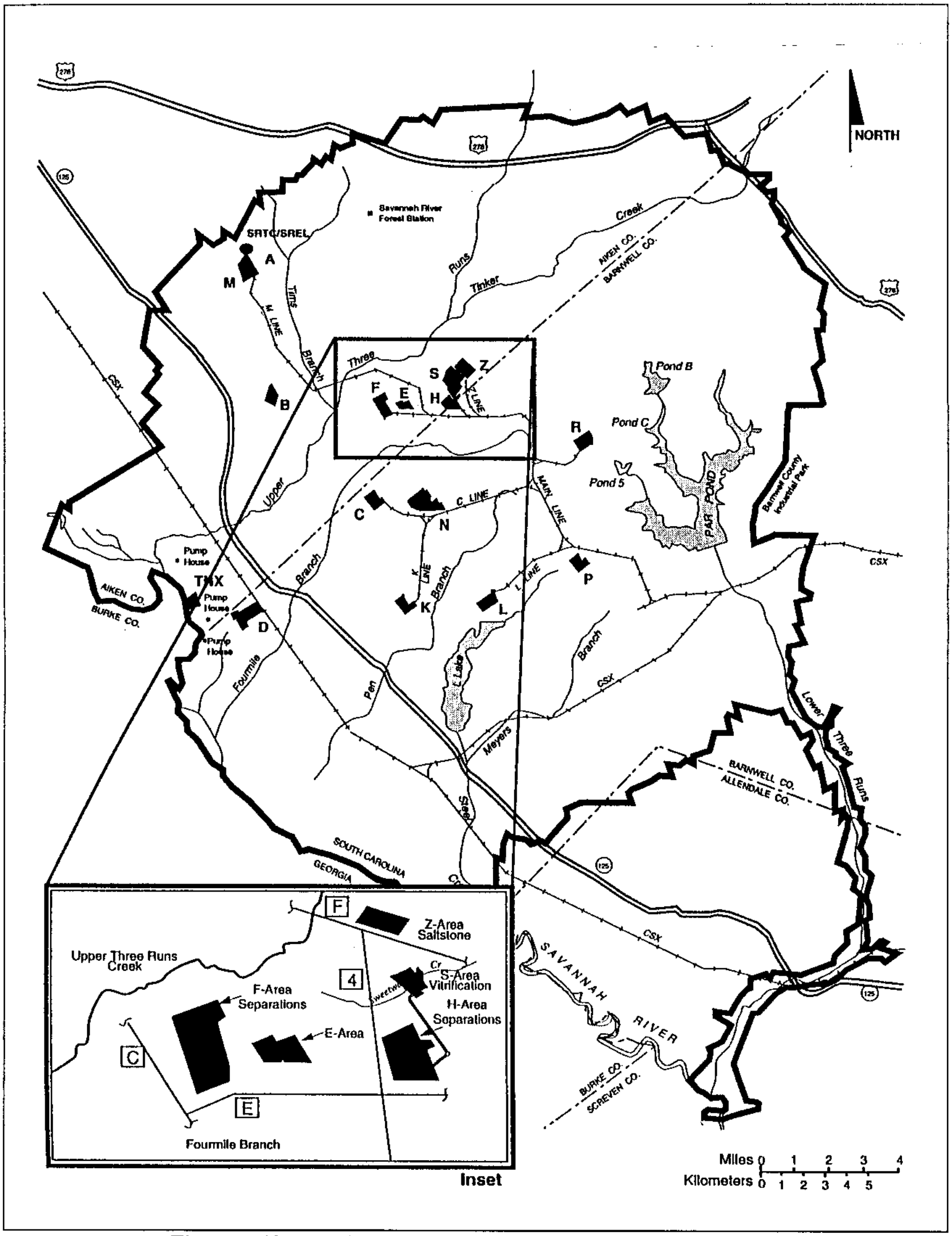


Figure F-13 Location of Principal Savannah River Site Facilities

The primary Savannah River Site facilities were used for the production of nuclear materials. Currently, the production reactor facilities are not operating and are in either shutdown or standby mode. Several large waste management projects are now underway at the site, including the Defense Waste Processing Facility for the vitrification of high-level waste.

F.1.3.1.1 Spent Nuclear Fuel Activities at the Savannah River Site

The Savannah River Site currently stores approximately 201 MTHM of spent nuclear fuel (DOE, 1995g), or approximately 7 percent of the DOE total, including the following:

- 184.4 MTHM of aluminum-based spent nuclear fuel, including plutonium target material,
- 4.6 MTHM of commercial spent nuclear fuel (zircaloy-clad),
- 11.9 MTHM of test and experimental reactor, zircaloy-clad fuel, and
- 5.4 MTHM of test and experimental reactor, stainless steel-clad fuel.

This fuel is stored in several basins onsite. The F- and H-Area Canyons are the processing and separations facilities at the Savannah River Site, and each has a small associated wet storage basin. Three reactor disassembly basins (K, L, and P) contain the reactor fuel and target materials. A fourth reactor disassembly basin (C) currently is the only basin without security upgrades necessary for any storage activities. These basins consist of unlined concrete with inadequate water purification equipment for extended storage of aluminum-clad spent nuclear fuels. These reactor basins were built in the 1950s and were not intended for the long-term storage ("years") of radioactive materials. Furthermore, poor water chemistry has corroded some of the spent nuclear fuel in the K- and L-Reactor disassembly basins, resulting in the release of fissile materials to the pool water. Also, these reactor basins are not seismically qualified and lack modern earthquake resistant features. Ongoing facility upgrades of the L-Reactor disassembly basin are intended to correct the conditions of the basin. Deionization of the basin has lowered the conductivity to acceptable levels for corrosion control. Lower conductivity would greatly reduce the probability of new corrosion and reduce the rate of progression of existing corrosion. The control of the conductivity after the completion of the deionization would be accomplished using the Disassembly Basin Upgrade Project which was initiated to address near term activities and vulnerabilities associated with storing fuel in the L-Reactor disassembly basin. With the upgrades to be completed by mid-1996 (Miller et al., 1995), the L-Reactor basin can be expected to safely store spent nuclear fuel for as long as 10 to 20 years. These upgrades include the following:

- A continuous on-line deionization system to improve water chemistry. The continuous deionization system will lower and control the conductivity levels of the basin thereby minimizing corrosion. The continuous deionizer system also removes ionic radionuclide concentrations, specifically Cesium-137.
- A makeup water deionizer to improve the quality of makeup water supplied to the basin. This action will mitigate any additional load on the continuous deionization system.
- New equipment and systems for alternative packaging and removal of waste.

A Basis for Interim Operation document for the L-Reactor in cold standby conditions was prepared by the Westinghouse Savannah River Company (WSRC, 1995b). The Basis for Interim Operation addressed the effects of process events on the facility worker and the effects of process and natural phenomena hazards events on the public and the environment. The Basis for Interim Operation document concluded that the facility could continue to operate within the safety envelope, identified in the Basis for Interim Operation, without undue risk to the public or the environment.

The RBOF is the other major facility for spent nuclear fuel storage. The RBOF is more suitable than the reactor basins because it is lined (epoxy sides, stainless steel bottom) and has a water purification system. The BNFP, after refurbishing, would be suitable for foreign research reactor spent nuclear fuel storage because it is fully lined with stainless steel, has water purification systems, and has active heat removal systems. Major spent nuclear fuel storage facilities are summarized in Table F-10.

Table F-10 Major Savannah River Site Spent Nuclear Fuel Storage Facilities

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel Elements*	*Access*
105-K Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-L Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-C Disassembly Basin	Basin Dimensions: 39.6 x 58.2 x 5.2m (130'W x 191'L x ~17'D) Basin Water: 13.6 million l (3.6 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-P Disassembly Basin	Basin Dimensions: 55.5 x 68.2 x 5.2m (182'W x 223'L x ~17'D) Basin Water: 18.2 million l (4.8 million gal)	None initially 20,000 after upgrades	Truck/Rail
RBOF (244-H)	Basin 1: 8.2 x 12.1 x 6.7m depth over two-thirds of floor space 8.8m depth over one-third of area Basin 2: 8.2 x 3.9 x 8.8m depth Basin Water: 1.7 million l (450,000 gal)	~1000 initially, plus 1,425 after rearranging[b]	Truck/Rail
BNFP[a]	Several Pools: Main Pool: 14.6 x 14.6 x 9.8m (48'L x 48' x 32'D) Basin Water: 2.1 million l (550,000 gal)	None initially 25,000 after acquisition and reactivation[b]	Truck/Rail

[a] *Discussed in more detail in Section F.1.3.1.3; rail spur not currently active but would be included in reactivation.*

[b] *Difference in capacity between RBOF and BNFP is due to greater pool depth of BNFP and different fuel packing density assumptions for the two facilities.*

F.1.3.1.2 Spent Nuclear Fuel Storage Facilities Available for Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site

The RBOF is the principal facility applicable for foreign research reactor spent nuclear fuel. This basin has been operating and receiving spent nuclear fuel, including foreign research reactor spent nuclear fuel, since 1964, and is located in H-Area, near the center of the Savannah River Site. The 1,393 m^2 (15,000 ft^2) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting canals hold approximately 1,893,000 l (500,000 gal) of water. Spent nuclear fuel elements arrive in lead-lined casks weighing from 22 to 64 metric tons (24 to 70 tons), which a crane lifts from a railroad car or a truck trailer and places in the unloading basin. About 30 percent of the fuels in the RBOF consist of uranium clad in stainless steel or zircaloy, which the Savannah River Site facilities cannot process without modifications. The RBOF is discussed in more detail in Section F.3.

In March 1995, the Savannah River Site estimated that the RBOF has the capacity for approximately an additional 1,000 spent nuclear fuel elements (O'Rear, 1995). However, the Savannah River Site has determined that 1,425 additional spaces can be made available by rearranging fuel in the pools and moving spent nuclear fuel to other storage areas, such as one of the reactor disassembly basins. If empty, the total RBOF capacity would be 6,500 foreign research reactor spent nuclear fuel elements.

F.1.3.1.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Savannah River Site for Foreign Research Reactor Spent Nuclear Fuel

The Savannah River Site is evaluating the use of several new planned or potential facilities for foreign research reactor spent nuclear fuel management. These include:

- a modular dry vault storage building,
- dry cask storage, or
- wet pool storage.

These technologies may require additional support facilities for such functions as: spent nuclear fuel examination, spent nuclear fuel characterization, cask loading and unloading, spent nuclear fuel repackaging, and cask maintenance. The Savannah River Site is also evaluating the use of one or more of the reactor disassembly basins for near-term wet storage of foreign research reactor spent nuclear fuel. These facilities are discussed in more detail in Section F.3.

The Savannah River Site is also evaluating the potential storage of spent nuclear fuel at the BNFP facility. Allied General Nuclear Services constructed a large reprocessing facility for commercial spent nuclear fuel in Barnwell, South Carolina, adjacent to the Savannah River Site (Fields, 1994; Matthews, 1994 and 1991; Taylor et al., 1994; Williams, 1994; WSRC, 1992a-d). This plant was never operated due to a change in Government policy, and was mothballed in the 1980's. The BNFP includes a wet fuel storage basin that is approximately twice the area and potentially has over four times the spent nuclear fuel capacity of the RBOF facility at the Savannah River Site. The wet storage basin is fully lined and seismically qualified and would be capable of storing all of the currently identified foreign research reactor spent nuclear fuel (Jackson, 1994). Facility acquisition, replacement of removed equipment, reactivation, installation of suitable storage racks, and checkout at the facility would be required prior to its use.

Figure F-14 displays the location of the BNFP in relation to the Savannah River Site. This land was originally part of the Savannah River Site. The BNFP site consists of approximately 680 hectares (ha) (1,680 acres).

Allied General Nuclear Services designates the fuel pool area of the plant as the "Fuel Receiving and Storage Stations." Considerable documentation exists for the facility, including the engineering designs, the Environmental Impact Statement (EIS), and the Final Safety Analysis Report submitted to the NRC. The pools and attendant cranes are fully seismically qualified structures. The pool section includes ion exchange systems for pool water purification and a separate radwaste system (solidification may need to be added). The section incorporates capabilities for receipt of either truck or railcarried casks. The main crane is rated at 122 metric tons (135 tons).

The Fuel Receiving and Storage Station facility is shown in Figure F-15 and was designed and constructed to receive, store, and handle spent (irradiated) light water reactor fuel. Spent nuclear fuel assemblies are received in shielded casks by either truck or rail. The assemblies are unloaded underwater and stored underwater to provide cooling and shielding. Stored fuel can be remotely transferred to the adjacent Remote Process Cell and Remote Maintenance and Scrap Cell for mechanical processing. After fuel assemblies are unloaded from the shielded casks, the empty casks are prepared for return shipment.

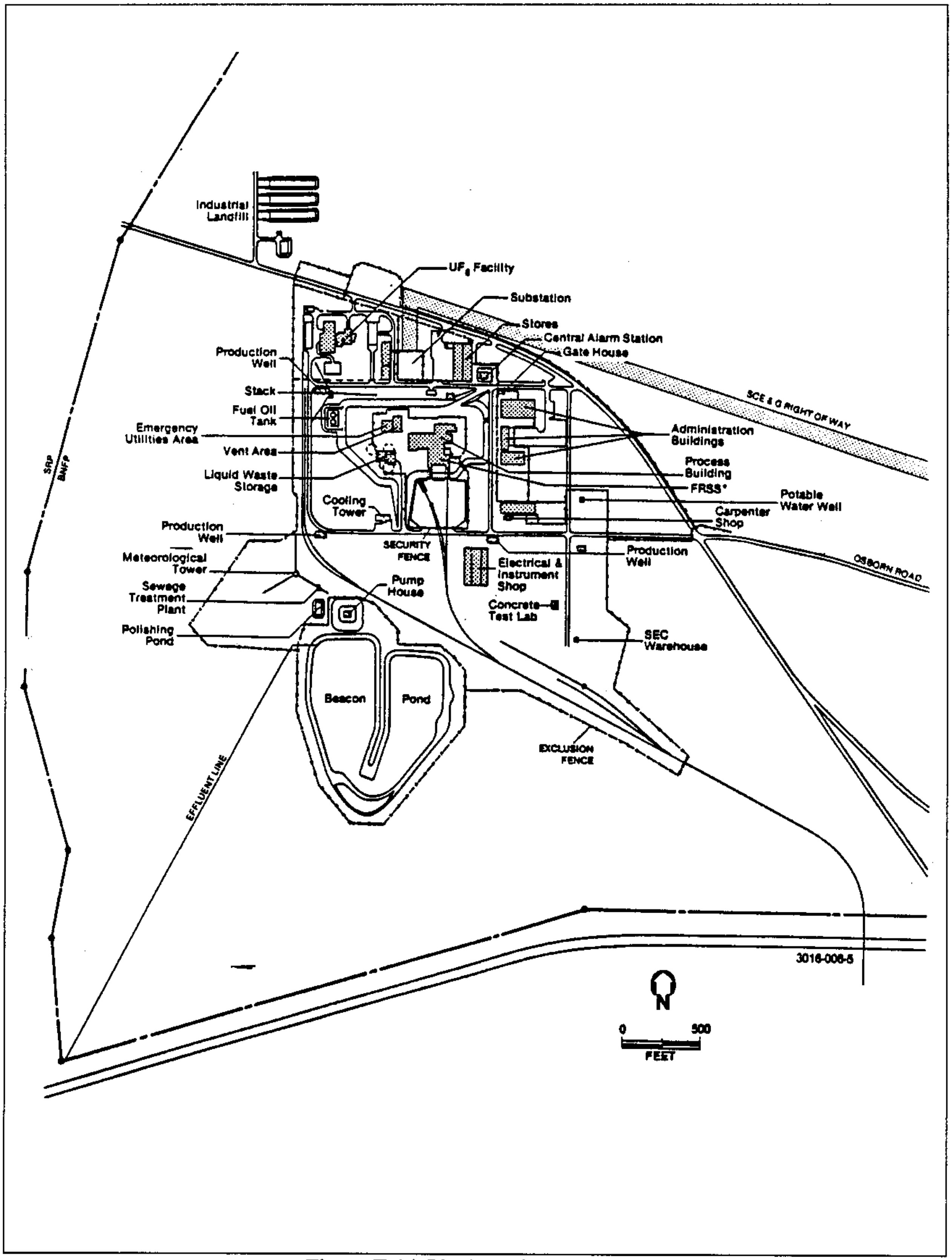


Figure F-14 Plot Plan for the BNFP

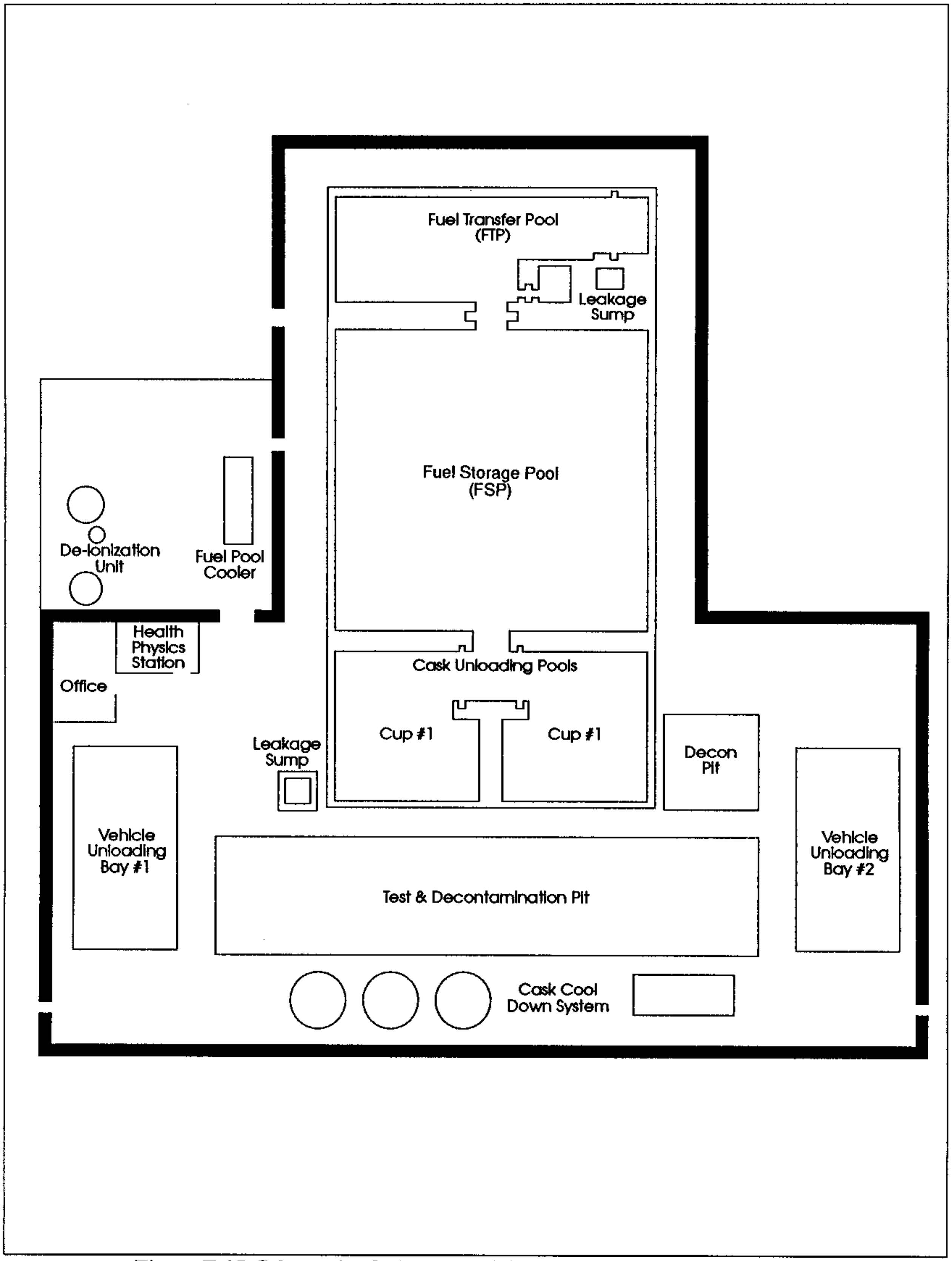


Figure F-15 Schematic of a Fuel Receiving and Storage Station at BNFP

The following areas of the Fuel Receiving and Storage Station are safety class structures:

- pool concrete structure,
- pool and crane column foundations,
- embedments for the fuel storage racks,
- crane rails, rail supports, and restrainer bars which retain the cranes on their rails and prevent their falling into the pools,
- cask barrier beams and embedments,
- energy absorbing pads in the Cask Unloading Pools,
- emergency water supply line, and
- Fuel Receiving and Storage Station walls to the 7.6 m (25 ft) level above grade. Clean spent nuclear fuel casks are moved to the Fuel Receiving and Storage Station water pool area. This area is divided into six pools consisting of two Cask Unloading Pools, one Fuel Storage Pool, one Failed Fuel Pool, one Fuel Transfer Pool, and an examination cell/pool.

Water shielding of 3.7 m (12 ft) is provided in the Fuel Receiving and Storage Station pools. This limits surface dose rates to a calculated 0.08 mrem/hr, assuming design basis Light Water Reactor fuel, and permitting at least 40 hours per week working time for an operator. Handling systems are designed with special limit switches and mechanical stops to prevent raising fuel higher than the design depth of the shielding water.

The water in the five pools of the Fuel Receiving and Storage Station is channeled and treated to promote maximum clarity, to control temperature, and to minimize corrosion and radioactivity. This is accomplished by continuous filtration through 95 percent efficient 5 micron pore size filter elements, cooling in heat exchangers to hold the pool water temperature below 41°C (105°F), and demineralization.

Demineralizing water treatment is designed to maintain radioactivity levels below 0.0005 μCi/ml. Pool water is pumped from the Fuel Storage Pool at 7,570 l/min (2,000 gal/min), directed through the heat exchangers, and returned to the Fuel Storage Pool. A second stream is pumped at 1,135 l/min (300 gal/min) from a pool and is filtered. After filtration, one-half of this stream is treated by ion exchange. The combined filtered and purified solution is then returned to the Fuel Storage Pool. The pool piping system is arranged so that the cleanup stream can be removed from or returned to any of the pool areas, permitting cleanup of contaminated water.

The cooling system is designed to remove heat at a rate of 4,000 kilowatts (14 million Btu/hr). The cooling capacity can be increased by expanding the capacity of the heat exchanger system in the Fuel Receiving and Storage Station. The estimated life for the structure is 50 years (Fields, 1994; Matthews, 1994; Taylor et al., 1994).

The Fuel Receiving and Storage Station has the following six pools:

- two cask pools, each 18.3 m (60 ft) deep,
- failed fuel pool (for degraded fuel),
- fuel transfer pool, 18.3 m (60 ft) deep,

- examination cell/pool, and
- main pool, 14.6 m x 14.6 m x 9.8 m deep (48 ft by 48 ft by 32 ft deep).

All of the pools are lined with stainless steel and are designed to maintain a minimum of 3.7 m (12 ft) of water above the fuel for shielding. The pools include detectors and flow channels for managing potential leaks. The original capacity of the main pool was 400 MTHM. Various analyses have been performed to increase this capacity to the 1,200 to 2,000 MTHM range with reracking and other arrangements. It has been estimated that maximum wet storage corresponds to approximately 5,200 Pressurized Water Reactor assemblies (Taylor et al., 1994). For foreign research reactor spent nuclear fuel, this would correspond to over 25,000 elements; and, thus, as noted previously, the BNFP could accommodate all of the fuel.

The environmental impacts of spent nuclear fuel storage at the BNFP have also been analyzed for between 360 and 5,000 MTHM of commercial fuel (Taylor et al., 1994). The results were:

- Dose commitments to the 80 km (50 mi) population were estimated to be 0.067 person-rem and 0.071 person-rem for 15- and 25-year storage periods, respectively.
- The worst accident would result in a dose commitment of 1 mrem total body, 6 mrem thyroid, and 100 mrem skin to an exposed individual located at the eastern boundary of the site.

These analyses were based upon commercial spent nuclear fuel, but should bound the consequences of foreign research reactor spent nuclear fuel storage at the BNFP. Potential impacts are discussed in more detail in Section F.4.

The BNFP site consists of some 680 ha (1,680 acres), bounded on three sides by the Savannah River Site. Preliminary walkthroughs and analyses by the Savannah River Site indicate the facility is in good condition, and principally needs a main transformer for power supply. The Savannah River Site has estimated a cost of $50 million (Matthews, 1991; WSRC, 1992a-d). Actual acquisition and reactivation costs are claimed to be as low as $25 million (Matthews, 1994; WSRC, 1992a-d). This facility, however, would not be available immediately to receive the foreign research reactor spent nuclear fuel.

Figure F-16 displays the foreign research reactor spent nuclear fuel storage capacity versus time for the Savannah River Site. Clearly, the Savannah River Site can accommodate foreign research reactor spent nuclear fuel at existing facilities supplemented by dry storage, modified reactor disassembly basins, or the potential use of the BNFP. The reactor basins could be used to store the non-aluminum-based spent nuclear fuel currently in the RBOF because the poorer water quality in the basins would not cause additional corrosion for this other fuel that is not aluminum based. Recent improvements in reactor basin water chemistry control have resulted in a substantial decrease in the potential for corrosion of aluminum-clad spent nuclear fuels.

F.1.3.2 Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory has several reactors and critical assemblies operating and also possesses several reactors that are either in standby or shutdown and awaiting decommissioning. From 1953 until 1992, the Idaho National Engineering Laboratory was responsible for processing and recovering highly-enriched uranium (HEU) from naval reactors. The Idaho National Engineering Laboratory discontinued processing spent nuclear fuel in 1992. Consequently, the Idaho National Engineering Laboratory has spent nuclear fuel facilities, spent nuclear fuel in storage, and spent nuclear fuel from

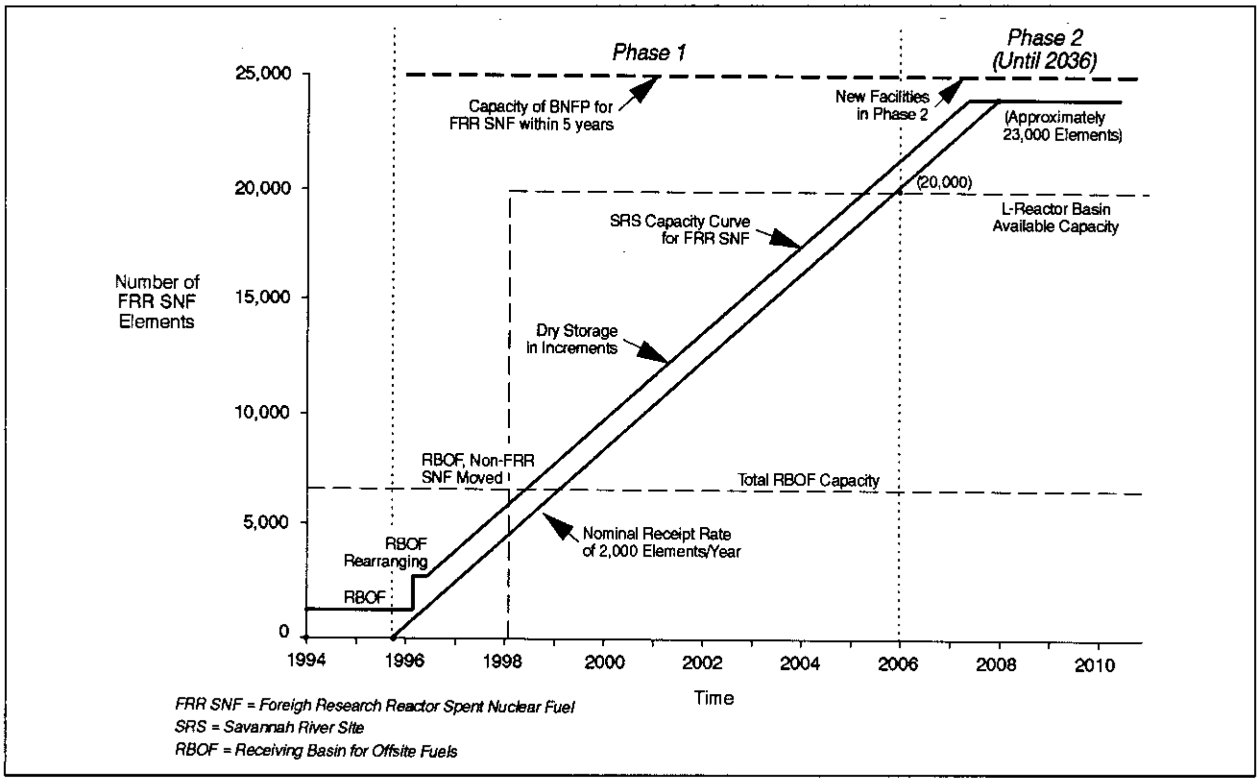


Figure F-16 Foreign Research Reactor Spent Nuclear Fuel Storage at the Savannah River Site

current operations. The Idaho National Engineering Laboratory site map with spent nuclear fuel facilities is shown in Figure F-17.

F.1.3.2.1 Spent Nuclear Fuel Activities at the Idaho National Engineering Laboratory

Six major facility areas at the Idaho National Engineering Laboratory store spent nuclear fuel:

- ICPP,
- Test Area North,
- Power Burst Facility,
- Test Reactor Area,
- Argonne National Laboratory-West, and
- Naval Reactors Facility.

A description of each major facility area and its spent nuclear fuel storage activities is presented below.

F.1.3.2.2 Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory

Spent nuclear fuel at the Idaho National Engineering Laboratory is sorted in a variety of dry and wet configurations. The total amount of spent nuclear fuel at the Idaho National Engineering Laboratory

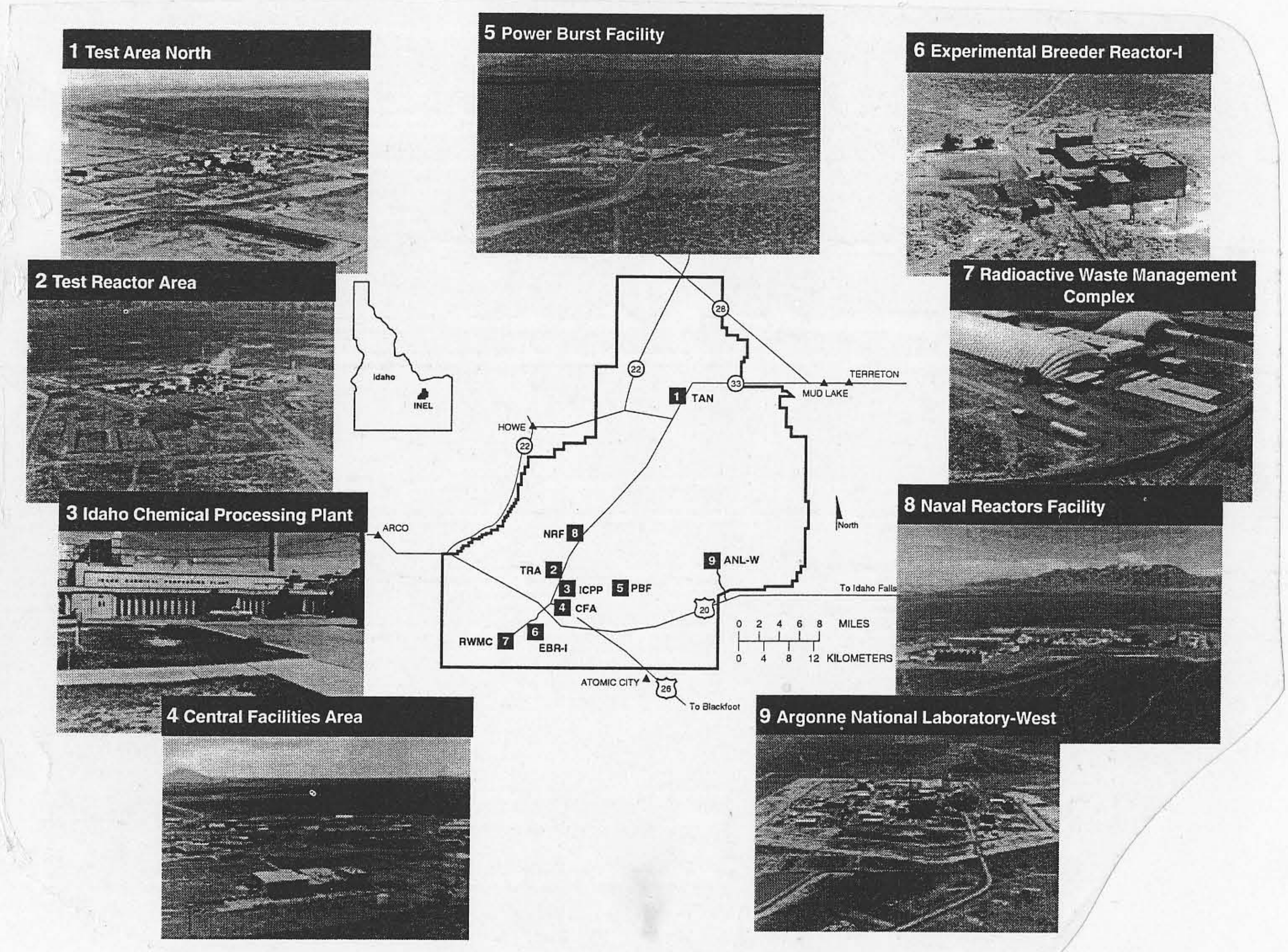


Figure F-17 Location of Spent Nuclear Fuel Storage and Handling Facilities at the Idaho National Engineering Laboratory accounts for about 10 percent (by weight of heavy metal) of the spent nuclear fuel in the DOE complex (DOE, 1995g).

Table F-11 lists the primary spent nuclear fuel storage facilities, including the type of storage configuration, capacity for foreign research reactor spent nuclear fuel receipts, and accessibility. The number, variety, and location of the wet and dry configurations currently in use at the Idaho National Engineering Laboratory are largely the result of the different purposes for the facilities (e.g., at-reactor storage, storage research and development, reprocessing, and fuel research and development). The condition of the spent nuclear fuel in storage is generally good, with the notable exception of minor amounts of fuel in the Underwater Fuel Storage Facility at the ICPP-603.

The ICPP has received spent nuclear fuel from many onsite and offsite reactors (including foreign research reactor spent nuclear fuel) for reprocessing. Reprocessing for recovery of HEU materials was ceased in 1992. The ICPP now has the mission of managing its current spent nuclear fuel inventory and assigned new spent nuclear fuel receipts, development of technologies in support of dispositioning the spent nuclear fuel, and eventually packaging the material for shipment to a repository. The ICPP stores virtually all types of spent nuclear fuel except production reactor fuel (i.e., fuel from the Hanford Site and the Savannah River Site production reactors). It stores nonproduction reactor aluminum, stainless steel, zirconium, and graphite-clad spent nuclear fuel and uses both wet and dry storage configurations. The ICPP facilities have experience and some capacity for foreign research reactor spent nuclear fuel storage. These are discussed in more detail in Section F.3.

The Test Area North has been a reactor testing facility and has received significant amounts of spent nuclear fuel for examination and testing purposes. This includes the commercial dry storage cask demonstration program and the Three Mile Island debris examination program. It has a very large hot cell and an adjacent underwater storage pool to support the testing programs. It also has a large hot shop where large pieces of equipment, such as transportation casks, have been reconfigured or maintained. At the current time, the Test Area North hot cell and pool have no future mission, but may be used by the U.S. Navy. If Test Area North is not used by the Navy, then the Test Area North hot cell and pool may have significant capacity for receipt of foreign research reactor spent nuclear fuel and for placing it into temporary underwater storage or dry storage casks.

Other storage areas such as the Power Burst Facility reactor canal and the MTR storage pool have limited storage capacities for receipt or storage of foreign research reactor spent nuclear fuel.

The Argonne National Laboratory-West facilities supported the Experimental Breeder Reactor program and also contain the Transient Reactor Test Facility, the Zero Power Physics Reactor, and the Neutron Radiography Reactor. Spent nuclear fuel storage facilities include an at-reactor molten sodium storage pool, in-process lag storage in the Hot Fuel Examination Facility and dry underground SILOs for spent fuel and wastes pending disposition. The Hot Fuel Examination Facility would be suitable for foreign research reactor spent nuclear fuel examination activities.

The Naval Reactors Facility is also located at the Idaho National Engineering Laboratory, but is not included in Table F-11 because of its sole purpose to support the Naval ship propulsion program. The Naval Reactors Facility includes the Expended Core Facility, which receives and examines Naval spent nuclear fuel to support fuel development and performance analyses. In addition, the Expended Core Facility removes structural support material from the Naval spent nuclear fuel before transfer of the fuel portion to the ICPP for reprocessing or interim storage.

Table F-11 Description of Existing Spent Nuclear Fuel Facilities at the Idaho National Engineering Laboratory

Facility	Description	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
ICPP-666 Underwater Fuel Storage Area	Water Storage Facility with 6 lined storage basins 9.4 m x 14.2 m by 9.4 m or 12.5 m deep (31 ft x 46.5 ft x 31 ft or 41 ft deep)	Temporary storage after reracking for 8,400 elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-603 Underwater Fuel Storage Area	Water Storage Facility with three basins of varying sizes, no sealant or liner	Not Available - facility is being shut down	Shipment by truck.
ICPP-603 Irradiated Fuel Storage Facilities	Dry Storage Facility with remote unloading area and vault storage with 636 0.5 x 3.4 m L (18 in x 11 ft long) containers	200 containers available for storage of 9,000 foreign research reactor elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-749 Underground Fuel Storage Area	Dry Storage Facility with 218 underground SILOs	Approximately 60 SILOs available following renovation of first generation SILOs; capacity for 3,600 elements after fiscal year 1998	Requires receipt into ICPP-666 or ICPP-603 IFSF and packaging and conditioning for dry storage.
Test Area North-607 Pool and Hot Cell	Water Storage Facility with adjacent remote hot cell	Approximately 56 m^2 (600 ft^2) of basin 7.3 m (24 ft) deep. Capacity for 4,000 elements after new rack installation.	Additional storage space available in hot cell. Shipment by truck, cask unloading in hot cell.
Test Reactor Area-620 Power Burst Facility	Small water storage pool adjacent to Power Burst Facility reactor	Minimal space available	Shipment by truck. Crane capacity inadequate for foreign research reactor casks.
Test Area North-607 Cask Storage Pad	Five commercial fuel storage casks on concrete pad	Easily expandable for more cask storage as necessary for foreign research reactor spent nuclear fuel shipments	Shipment by truck to hot cell where foreign research reactor spent nuclear fuel can be transferred to storage casks and moved to storage pad.
Test Reactor Area-603 MTR Pool	Water Storage Pool in basement of the MTR	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Test Reactor Area-660 ARMF and CFRMF Reactors and Canal	Swimming pool reactors with connecting canal	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Argonne National Laboratory-West Hot Fuel Examination Facility	Large two room hot cell facility for fuel examinations with argon atmosphere	Minimal space available without extensive removal of examination equipment	Shipment by truck access to hot cell limited to special designed small transfer cask.
Argonne National Laboratory-West Radioactive Scrap and Waste Facility	1,200 vertical steel-lined underground dry storage wells	About 500 wells are not used	Access requires transfer through Hot Fuel Examination Facility.

F.1.3.2.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory for Foreign Research Reactor Spent Nuclear Fuel

The main focus of near-term activities is the accurate quantification and characterization of DOE-owned spent nuclear fuel, identification of spent nuclear fuel management facilities and their conditions, identification of safe interim storage for existing and new spent nuclear fuel, and identification of technologies and requirements to place DOE spent nuclear fuel in safe interim storage. Long-term activities include the development of final waste acceptance criteria requirements and stabilization technologies for alternate fuel disposition, construction of facilities to stabilize fuel to meet waste disposal requirements, processing of the fuel to a final waste form, and transportation of the waste form for disposition (discussed in more detail in Section F.3). As shown in Figure F-18, the Idaho National Engineering Laboratory has sufficient capacity for foreign research reactor spent nuclear fuel if the existing facilities are supplemented by dry casks.

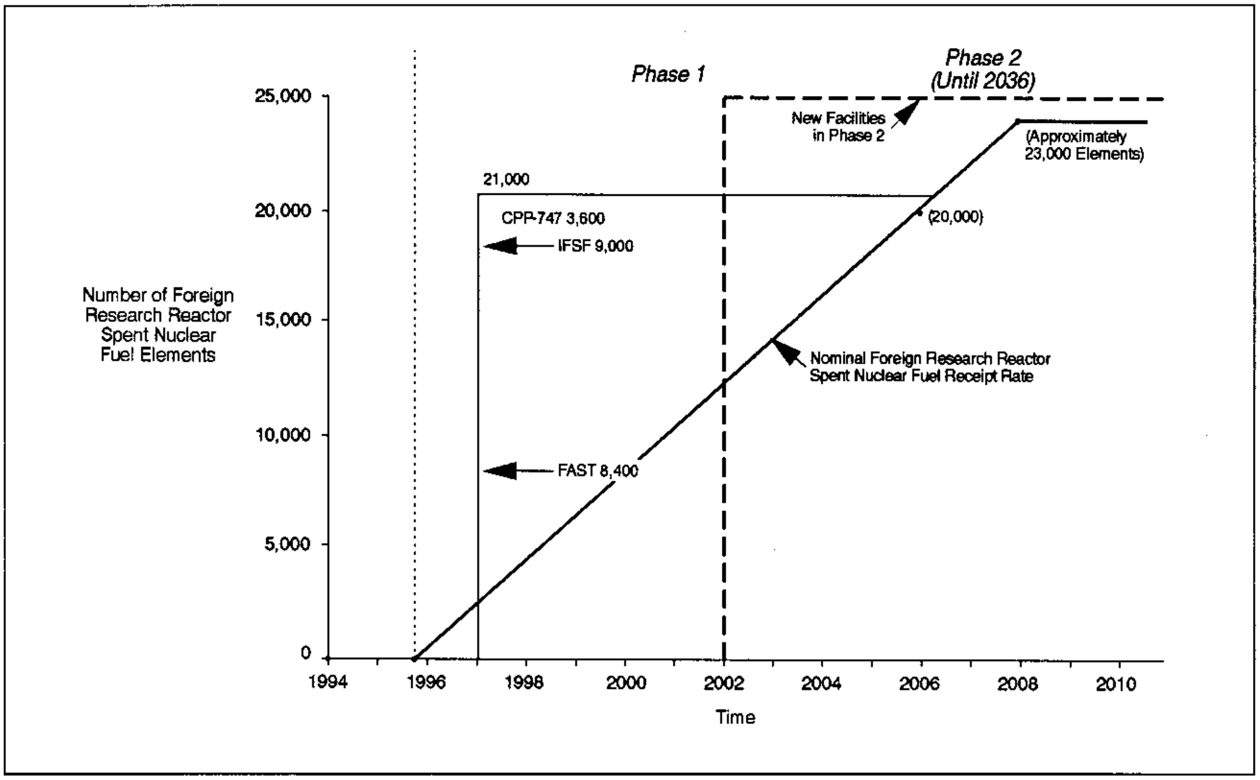


Figure F-18 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Idaho National Engineering Laboratory

F.1.3.3 Hanford Site

The Hanford Site lies within the semi-arid Pasco Basin of the Columbia Plateau in southeastern Washington State (DOE, 1995g). The Hanford Site occupies an area of around 1,450 km^2 (560 mi^2) north of the confluence of the Yakima and Columbia Rivers. Only about six percent of the site has been disturbed in the process of special nuclear materials production for national defense reprocessing and used for DOE purposes, such as nuclear materials production, processing, research and development, and waste management. The Hanford Site facilities include nine shutdown production reactors and several smaller research reactors. Several processing and product finishing facilities are located on the site, but are not currently operating and will not likely operate in the future. Currently, the principal mission of the site is environmental management and includes:

- decontamination and decommissioning of surplus facilities,
- environmental restoration of over 1,500 waste management units and 4 groundwater contamination plumes,

- waste management, including new processing facilities and retrievable disposal, and
- research and development into energy, environmental, and waste management technologies.

A Tri-Party Agreement between DOE, the U.S. Environmental Protection Agency, and the State of Washington provides milestones and guidance for these activities at the Hanford Site. Current schedules use a 2030 date for the completion of most of the restoration activities at the site. A map of the Hanford Site that shows spent nuclear fuel facilities is presented in Figure F-19. Existing spent nuclear fuel facilities are listed in Table F-12.

F.1.3.3.1 Spent Nuclear Fuel Activities at the Hanford Site

The following spent nuclear fuel types and their associated facilities are at the Hanford Site:

- *N Reactor Spent Nuclear Fuel:* This is zircaloy-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (1,146 and 954 MTHM, respectively), and exposed to air in the plutonium-uranium extraction dissolver cells A, B, and C (0.3 MTHM).
- *Single-Pass Reactor Spent Nuclear Fuel:* This is aluminum-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (0.4 and 0.1 MTHM, respectively), and stored in water in the plutonium-uranium extraction basin (approximately 2.9 MTHM).
- *Fast Flux Test Facility Spent Nuclear Fuel:* This consists of stainless steel-clad fuel stored in liquid sodium at the Fast Flux Test Facility, comprised mainly of a uranium/plutonium oxide fuel, but with some carbide, metallic, and nitride fuel elements (in all, fuel from 329 assemblies of spent nuclear fuel).
- *Shippingport Core II Spent Nuclear Fuel:* These assemblies are zircaloy-clad uranium dioxide fuel, and are stored in the T-Plant Canyon, Pool Cell 4.
- *Miscellaneous Commercial and Experimental Spent Nuclear Fuel:* This includes primarily zircaloy-clad uranium dioxide fuel stored in air, but does include some Test, Research, Isotope, General Atomic (TRIGA) reactor hydride spent nuclear fuel stored in water and aluminum-clad, uranium-aluminum metallic fuel stored in air. These are principally stored in the 300-Area at Hanford Site.

F.1.3.3.2 Spent Nuclear Fuel Storage Facilities at the Hanford Site

The Hanford Site spent nuclear fuel storage facilities are principally based upon wet methods. Table F-12 provides a brief summary of these facilities. The age, condition, available capacity of these facilities, and the Tri-Party Agreement milestones generally prevent the use of the existing facilities for storage of foreign research reactor spent nuclear fuel. It is extremely unlikely that significant processing activities on spent nuclear fuel will occur in the near future, and thus, new facilities would be required for foreign research reactor spent nuclear fuel management at the Hanford Site.

Two spent nuclear fuel EIS documents address the environmental impacts from spent nuclear fuel management at the Hanford Site. The first is the DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS (SNF&INEL Final EIS) (DOE, 1995g), the Record of Decision of which was issued on May 30, 1995, that in general specifies spent nuclear fuel management throughout DOE; and in particular

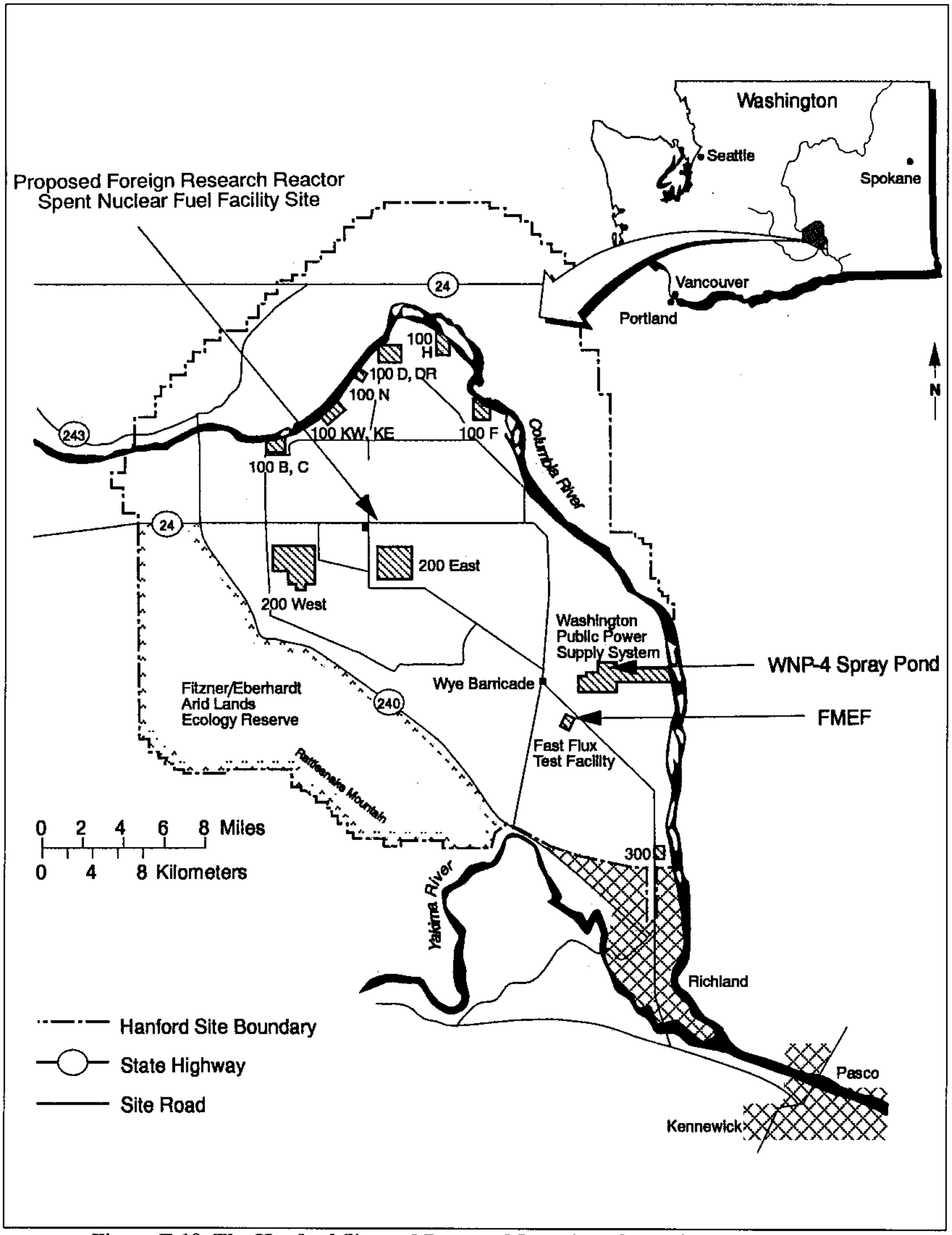


Figure F-19 The Hanford Site and Proposed Location of New Spent Nuclear Fuel Storage Facility

Table F-12 Description of Existing Spent Nuclear Fuel Facilities at Hanford Site

Facility	Description	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
105-KE Basin[a]	Water storage pool; 38 m x 20 m x 6 m deep, concrete walls and floor, no sealant or liner	75% - 100% full	By rail, 27 metric tons crane, fairly restrictive
105-KW Basin[a]	Water storage pool, 38 m x 20 m x 6 m deep, concrete walls and floor, epoxy sealant, no liner	75% full; no space for foreign research reactor spent nuclear fuel	By rail, 27 metric tons crane, fairly restrictive
T Plant: Cell 4	Water storage pool, 4 m x 8.4 m x 5.8 m deep	50% full; no space for foreign research reactor spent nuclear fuel	By rail or truck All fuel handling remote
PUREX Plant: East end of 202A Bldg, plus Dissolver Cells A, B, and C[b]	Water storage pool, 9.5 m x 6.1 m x 5.2 m deep, Dissolver Cell sizes vary	No additional capacity	Shipment by rail 36 metric tons crane
Plutonium Finishing Plant: 2736-ZB Bldg.	Dry storage in 208 L[c]	No additional capacity	Shipment by truck
FFTF: Reactor in-vessel storage, interim decay storage, and fuel storage facility locations[b]	Liquid sodium pool storage (fuel storage facility is separate from reactor containment building, with limit of kilowatts/assembly)	More than 75% full; no space for foreign research reactor spent nuclear fuel	By truck 91 metric tons Crane
200 Area LL Burial Grounds: 218-W-4C Trenches 1 and 7; and 218-W-3A Trenches 8 and S6	Dry, retrievable storage, 13 lead-lined, concrete-filled 208 liter drums, soil covered, 22 concrete casks (1.66 m x 1.66 m x 1.22 m or 1.92 m high), soil covered, 39 EBR II casks (1.5 m high x 0.4 m diameter), soil covered; 1 Zircaloy Hull Container (152 cm long x 76 cm diameter)	Large additional capacity; not suitable for foreign research reactor spent nuclear fuel	By truck
308 Building Annex: Neutron Radiography Facility[b]	Built in late 1970s water storage pool, 2.8 m diameter x 6 m deep	Small additional capacity	Truck shipments 4.5 metric tons crane
324 Building: B and D Cells	Dry storage in air, B Cell: 6.7 m x 7.6 m x 9.3 m high (spent nuclear fuel uses 10% of floor space). D Cell: 4 m x 6.4 m x 5.2 m high (small part for fuel), thick concrete walls and floors with steel liners	Small additional capacity	Truck shipments only B Cell - 2.7 and 5.4 metric tons cranes; Airlock - 27 metric tons crane
325 Building: A and B Cells in 325A Radiochemical Facility; 325B Shielded Analytical Laboratory	Dry Storage in air 325A - 1.8 m x 2.1 m x 4.6 m high (typical cell) 325B - 1.7 m x 1.7 m floor area (typical cell)	Small additional capacity	Truck shipments only 325A - 27 metric tons crane 325B - 2.7 metric tons crane
327 Building: A-F and I Cells; Upper and Lower SERF; Dry Storage vault, EBR II cask, Large Basin	Dry storage in air, except for water in small basin; variety of cell sizes, but storage only for fuel research	Small additional capacity	No direct rail Truck shipments 13.5 and 18 metric tons cranes

FFTF = Fast Flux Test Facility; EBR = Experimental Breeder Reactor; PUREX = Plutonium Uranium Extraction

[a] *If 105-KE Basin fuel is consolidated with 105-KW Basin fuel, 105-KE Basin would be shut down. The storage capacity of 105-KW Basin would be increased by replacing all of the storage racks to allow multi-tiered stacking of fuel canisters and by making minor facility modifications.*

[b] *Facility is being shut down.*

[c] *One 55 gal drum.*

specifies that Hanford generated spent nuclear fuel will remain in storage at Hanford pending decisions on ultimate disposition. The second, is the Management of Spent Nuclear Fuel from the K Basins at the Hanford Site Draft EIS (DOE, 1995d) which was issued for comment in October 1995. This EIS addresses the location and method of managing Hanford spent nuclear fuel for up to 40 years or until decisions on ultimate disposition are made.

New facilities would be required for storage of foreign research reactor spent nuclear fuel. However, there may be some economies of scale achieved from overlap with the other spent nuclear fuel activities at the Hanford Site.

F.1.3.3.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Hanford Site for Foreign Research Reactor Spent Nuclear Fuel

The Hanford Site has concluded that there are no existing facilities available and ready for accepting foreign research reactor spent nuclear fuel (Bergsman et al., 1994). Consequently, the Hanford Site proposes the following strategies:

- construction of a "generic" modular dry vault or dry cask storage facility,
- construction of a "generic" wet storage and handling facility,
- modification and completion of the Fuel Maintenance and Examination Facility (FMEF) (located at the Fast Flux Test Facility) as a modular dry vault storage facility, and
- acquisition, modification, and completion of the Washington Nuclear Plant-4 Spray Cooling Pond (at the Washington Public Power Supply System) as a wet storage facility.

These facilities and their potential applications to foreign research reactor spent nuclear fuel storage are discussed in detail in Section F.3.

Figure F-20 displays the Hanford Site capacity for foreign research reactor spent nuclear fuel storage. The Hanford Site is not considered capable of immediately accepting foreign research reactor spent nuclear fuel because of the required construction of new facilities. The Hanford Site would have sufficient capacity for foreign research reactor spent nuclear fuel storage after new facility construction.

F.1.3.4 Oak Ridge Reservation

The Oak Ridge Reservation is located on approximately 140 km^2 (54 mi^2) of federally owned land near Knoxville, TN (DOE, 1995g). There are three primary plant complexes within the Oak Ridge Reservation:

- *Y-12 Plant:* produces various materials used for national defense purposes,
- *K-25 Site (formerly called the Oak Ridge Gaseous Diffusion Plant):* originally used for uranium enrichment and now an environmental management site, and
- *Oak Ridge National Laboratory (also known as X-10):* research and development into nuclear energy and other energy technologies.

The Oak Ridge National Laboratory has operated several small reactors for research and isotope production and, of the Oak Ridge Reservation sites, is the most familiar with spent nuclear fuel

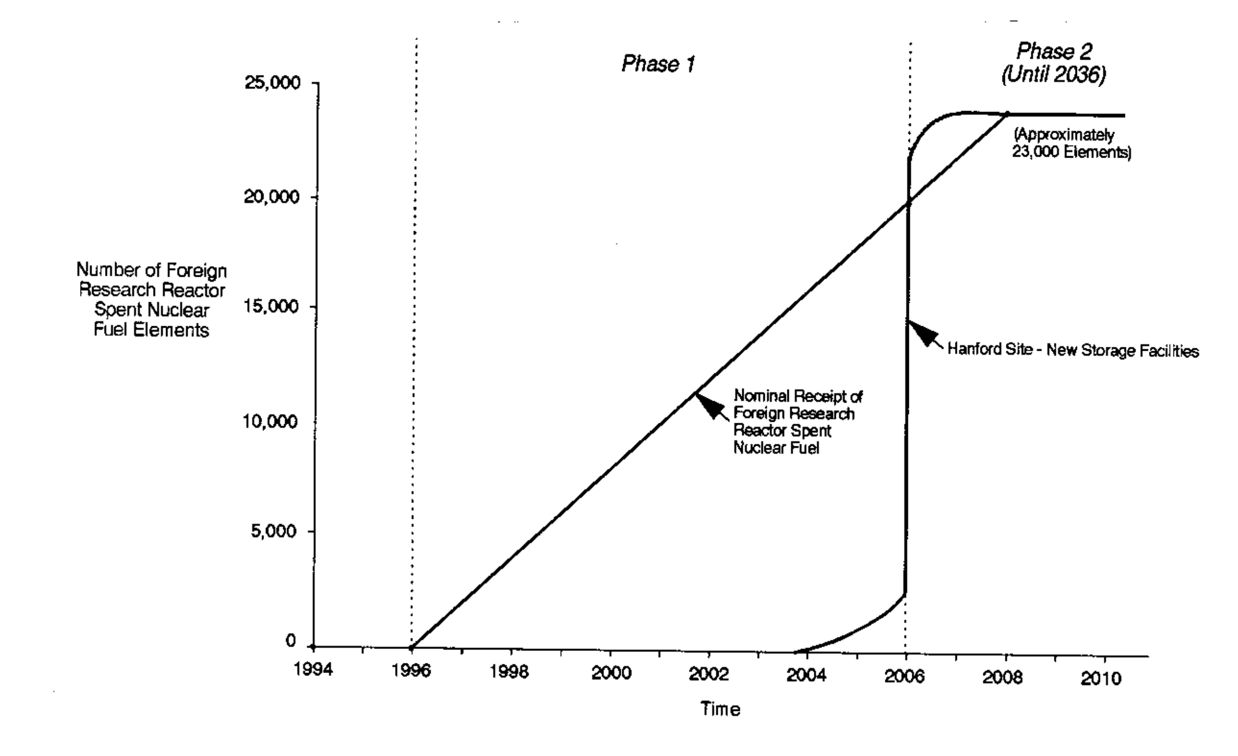


Figure F-20 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Hanford Site

requirements. A map of the Oak Ridge Reservation and its candidate sites for foreign research reactor spent nuclear fuel storage is presented in Figure F-21.

F.1.3.4.1 Spent Nuclear Fuel Activities at the Oak Ridge Reservation

Most Oak Ridge Reservation spent nuclear fuel activities occur at the Oak Ridge National Laboratory. The Oak Ridge National Laboratory has operated several small research reactors, all of which generate (or have generated) spent nuclear fuel. These reactors all have small fuel preparation and handling facilities associated with them ranging up to the single digit MTHM capacity. The spent nuclear fuel storage space is small, and most is either full or committed, with little excess capacity. The Oak Ridge National Laboratory also has hot cell and irradiated fuel examination facilities. Currently, only the High Flux Isotope Reactor is operating and generating spent nuclear fuel. More spent nuclear fuel facilities at Oak Ridge Reservation are presented in Table F-13.

F.1.3.4.2 Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation

The Oak Ridge Reservation stores spent nuclear fuel in several small facilities. Most of these facilities are old and are unlikely to meet modern building code and seismic standards. The spent nuclear fuel facilities include the following structures:

- *Building 3525 - Irradiated Fuels Examination Laboratory:* This two-story brick structure was constructed in 1963. It houses hot cells and contains small quantities of irradiated research reactor fuel in the form of samples and targets.

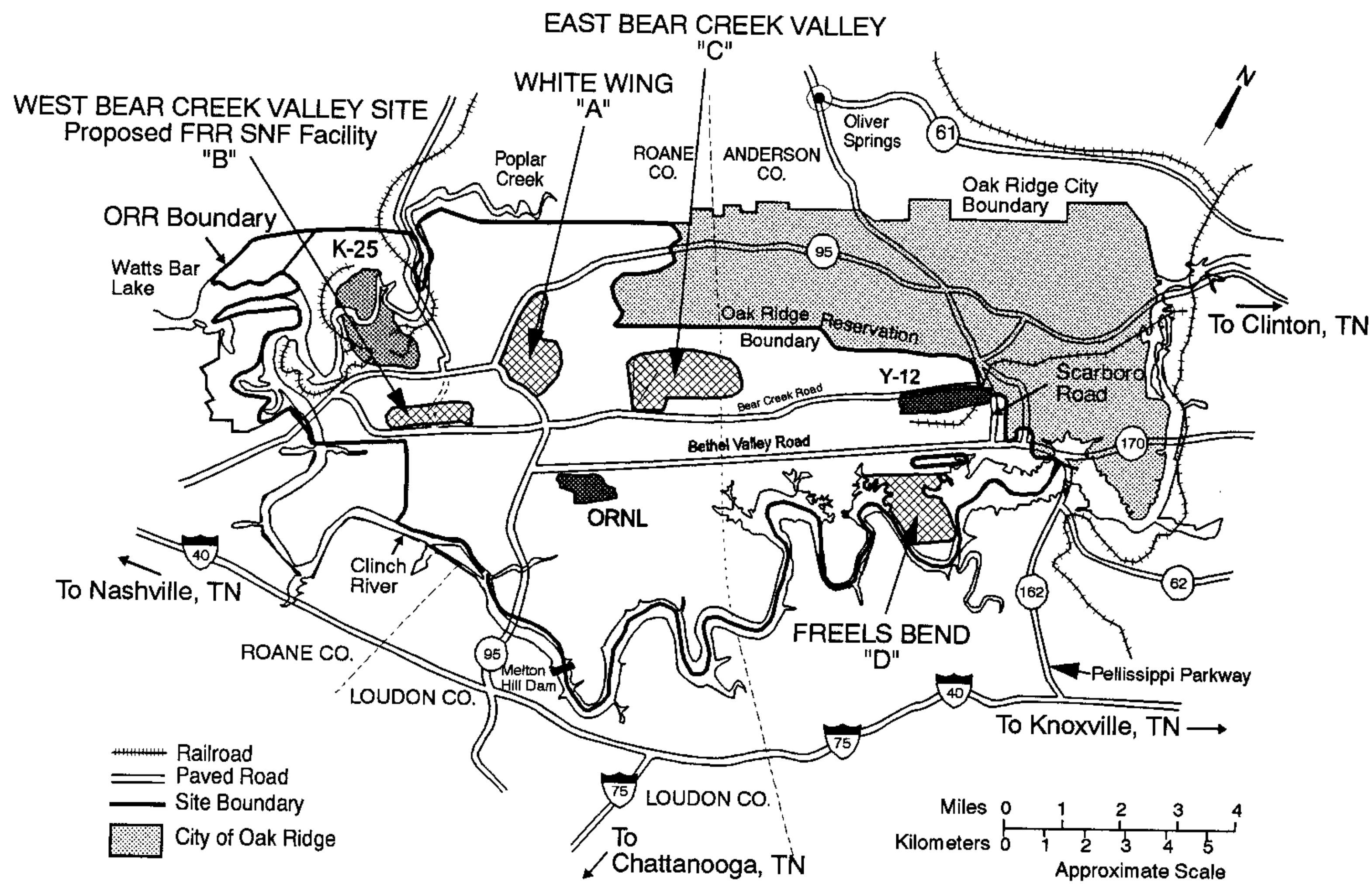


Figure F-21 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

Table F-13 Major Spent Nuclear Fuel Facilities at the Oak Ridge Reservation

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
Building 3525	Hot Cells	No, too small	Truck
Building 4501	Hot Cells	No, too small	Truck
Building 7827	Drywells	No space	Truck
Building 7920	Hot Cells	No, too small	Truck
Building 9720-5 (Y-12)	Warehouse	No, unirradiated fuel only	Truck
Other	Research Reactors	No, storage space near capacity	Truck

- *Building 4501 - High-Level Radiochemical Facility:* This facility dates from 1951 and contains hot cells for examining irradiated materials. This facility contains small quantities (several kg) of sectioned commercial fuel.

- *Building 7920 - Radiochemical Engineering Development Center:* This is a multi-purpose, hot cell facility for (relatively) large quantities of irradiated spent nuclear fuel. This facility supports target preparation and processing for the High Flux Isotope Reactor and contains samples and targets of research reactor spent nuclear fuel in dry storage.

- *Building 9720-5 (Y-12):* This is a large warehouse for storing and safeguarding unirradiated or low burnup HEU fuel. It currently contains around 0.2 MTHM.

- *Research Reactors:* There are five existing and one planned research reactor at the Oak Ridge Reservation. All of these reactors have small spent nuclear fuel storage basins nearby, and this capacity is essentially full. Only the High Flux Isotope Reactor is currently operating.

- *The Oak Ridge Reservation* also has several drywells such as Building 7827 and drum storage areas for irradiated fuel. Spent nuclear fuel would be relocated in accordance with actions of the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g).

None of these locations have any significant capacity for the potential quantities of foreign research reactor spent nuclear fuel.

F.1.3.4.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel

The Oak Ridge Reservation has plans for dry storage of spent nuclear fuel. This would be accomplished via a modular route at the High Flux Isotope Reactor location. This dry storage area could be extended almost indefinitely to accommodate the Oak Ridge Reservation's needs.

DOE is evaluating a spent nuclear fuel management complex for handling DOE spent nuclear fuel from other sites as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would include the following:

- Spent Nuclear Fuel Receiving and Canning Facility,

- Technology Development Facility,

- Interim Dry Storage Facility, and

- Expended Core Facility for Naval-type fuel similar to the one at Idaho National Engineering Laboratory.

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

The proposed site for the spent nuclear fuel facilities is located in the West Bear Creek Valley Area, in the western portion of the Oak Ridge Reservation site. This area of the Oak Ridge Reservation is currently in the Natural Areas land use category and is designated for future Waste Management land use. Land uses bordering on the Oak Ridge Reservation in this area are primarily agricultural farmland and commercial forest, with sparsely located residences (i.e., low population density).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not delineated.

Foreign research reactor spent nuclear fuel represents less than one percent of the DOE spent nuclear fuel quantities in terms of mass and, thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated. Figure F-22 summarizes foreign research reactor spent nuclear fuel capacity at the Oak Ridge Reservation. New facility construction would be required for foreign research reactor spent nuclear storage.

F.1.3.5 Nevada Test Site

The Nevada Test Site is located in the southeastern part of the State of Nevada, and is used as the on-continent site for nuclear weapons testing (DOE, 1995g). The Nevada Test Site encompasses approximately 3,500 km^2 (1,350 mi^2) of desert land, with flats, mesas, and mountain ridges (Figure F-23). Essentially no permanent surface waters exist, and the depth to groundwater routinely exceeds 330 m (1,000 ft). The Nellis Air Force Base Range surrounds the Nevada Test Site to the north, east, and west; and, with the Tonapah Test Range, provides a 24 to 104 km (15 to 65 mi) buffer zone between the Nevada Test Site and public lands. The Bureau of Land Management owns land on the southern and southwestern borders of the Nevada Test Site. Principal access to the site is via the town of Mercury, on the southeastern corner. Las Vegas is approximately 104 km (65 mi) from this corner of the Site.

Activities at the site have included nuclear weapons testing, nuclear reactor tests, nuclear rocket engine development, and waste management. Current activities include nuclear weapons-related activities (e.g., emergency search teams, arms control/verification, etc.), low-level waste/low-level mixed waste disposal, and site characterization for commercial spent nuclear fuel disposal.

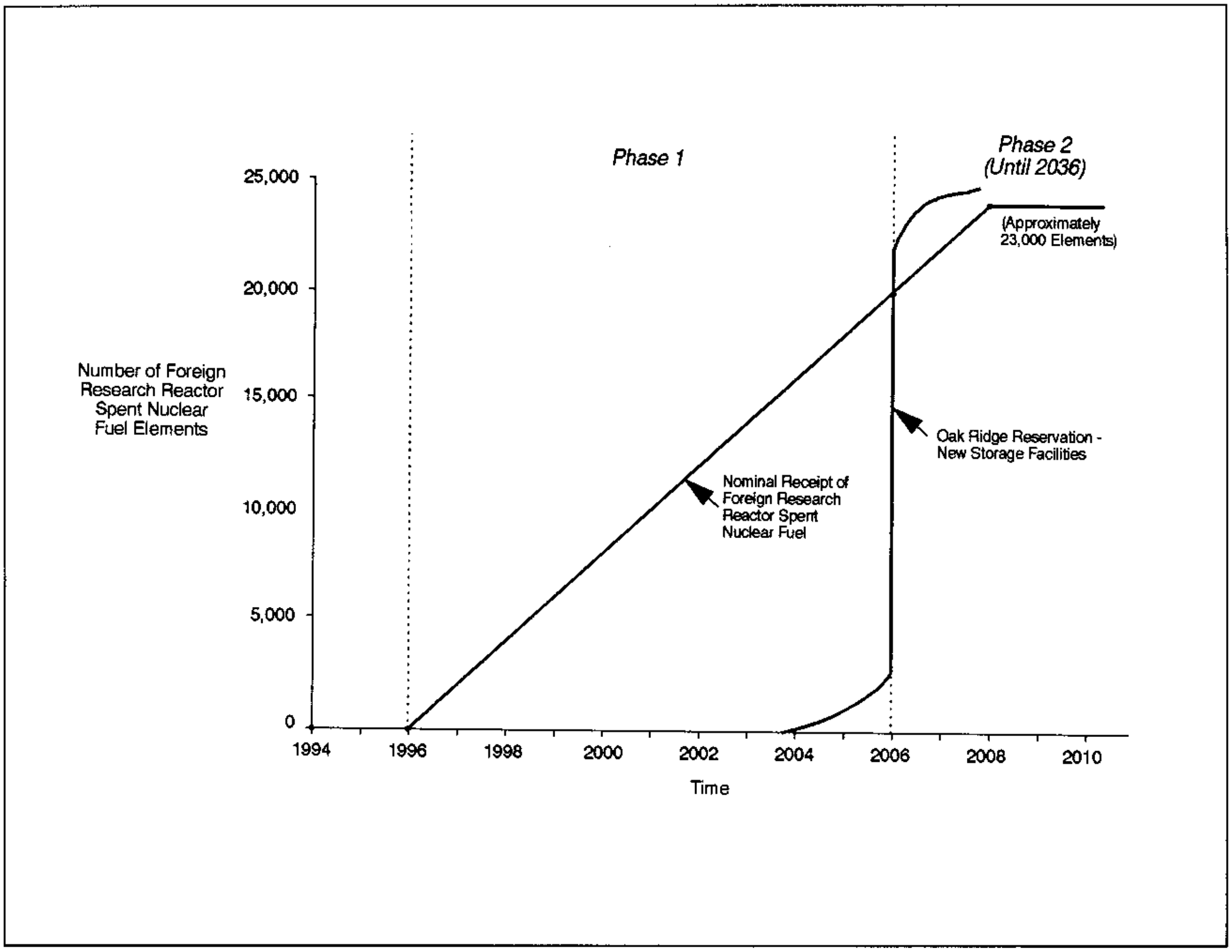


Figure F-22 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Oak Ridge Reservation

F.1.3.5.1 Spent Nuclear Fuel Activities at the Nevada Test Site

The Nevada Test Site has several existing facilities that could be useful for spent nuclear fuel management. These facilities were principally used for nuclear rocket engine development and are located at Jackass Flats, in a southern portion of the Nevada Test Site called the Nevada Research & Development Area (Cosimi, 1994; Chandler et al., 1992; Gertz, 1994; Hynes, 1994; Reed, 1994). The facilities include several large hot cell and fuel examination "shops," with large cranes and manipulators. At least two of these facilities appear to be ideally suited for handling and storing foreign research reactor spent nuclear fuel after relatively minor upgrades and refurbishments. Table F-14 summarizes the capabilities of these facilities for foreign research reactor spent nuclear fuel.

The Engine Maintenance and Disassembly (E-MAD) facility was originally constructed for the assembly and preparation of nuclear rocket engines for testing, refurbishment of activated engines, and disassembly and inspection of tested engines and components. The facility is designed for remote handling and examination of highly radioactive components. The building is a T-shaped, multi-storied structure, with overall dimensions of 85 x 107 m (280 ft x 350 ft) (Figure F-24). Numerous hot cells exist, with remote handling and transfer equipment, and the largest hot cell is 20 m wide x 45 m long x 23.5 m high (66 ft wide by 146 ft long and 77 ft high). Typically, 1.5 m (5 ft) thick concrete walls provide the shielding. Material transfer capabilities include several 36 metric tons (40 ton) cranes and a cask handling system of

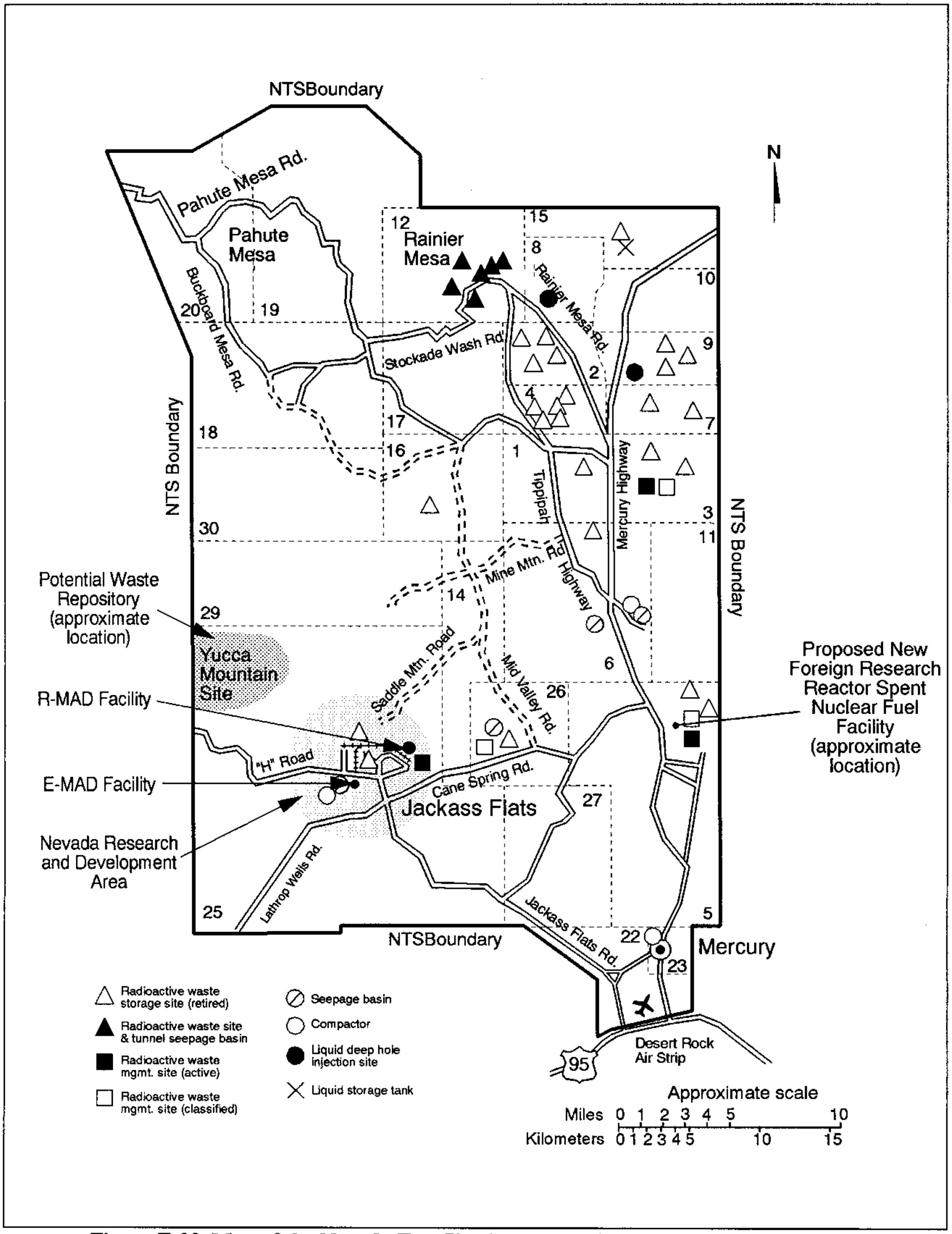


Figure F-23 Map of the Nevada Test Site Area 5 and Area 25 as Potential Foreign Research Reactor Spent Nuclear Fuel Storage Areas

Table F-14 Major Spent Nuclear Fuel-Capable Facilities at the Nevada Test Site

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
E-MAD	Large hot cell, with smaller hot cells. Main hot cell area is $895m^2$ (9,600 ft^2)	Yes, as either a vault or a staging facility for dry casks, after 1-2 years refurbishments, 25,000 elements	Truck
R-MAD	Large hot cell, with smaller hot cells. Main hot cell area is $223m^2$ (2,400 ft^2)	Yes, as a staging facility for dry casks or a small vault, after 1-3 years refurbishment, 25,000 elements	Truck

approximately 91 metric tons (100 tons) capacity. The heating, ventilation, and air conditioning systems for the hot cell areas maintain negative pressure and exhaust through High Efficiency Particulate Air filters.

The E-MAD facility is currently unused and last saw service during the 1980s for commercial spent nuclear fuel storage experiments (e.g., Climax Mine Project) (Gertz, 1994). Thirteen commercial spent nuclear fuel assemblies were tested in casks and drywells. The E-MAD facility was subsequently used to load transportation casks for shipment of the spent nuclear fuel to Idaho. Several of these spent nuclear fuel storage casks remain at the site (Hynes, 1994). The Los Alamos National Laboratory assessment (Chandler et al., 1992) considers the facility to require only minor upgrades and routine maintenance.

The Reactor Maintenance and Disassembly facility is located a short distance from the E-MAD facility. This facility contains two (contact) assembly bays and one remotely operated hot disassembly bay. The hot bay dimensions are 18 x 12 x 18 m (60 by 40 by 60 ft) high, with 1.8 m (6 ft) thick walls for shielding. A transfer system connects six hot cells to the hot disassembly bay. The Los Alamos National Laboratory assessment (Chandler et al., 1992) found the Reactor Maintenance and Disassembly facility to require a minor upgrade.

F.1.3.5.2 Spent Nuclear Fuel Storage Facilities at the Nevada Test Site

At the present time, the Nevada Test Site is not storing spent nuclear fuel. As noted, facilities in the Jackass Flats area have handled spent nuclear fuel in the past and could be adapted to accommodate foreign research reactor spent nuclear fuel and serve as the nucleus of a spent nuclear fuel storage facility. The E-MAD and Reactor Maintenance and Disassembly facilities appear to have sufficient size and design for accommodating all of the foreign research reactor spent nuclear fuel in a dry storage mode, either vault or cask, and for accomplishing any required transfer, examination, and canning operations.

F.1.3.5.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Nevada Test Site for Foreign Research Reactor Spent Nuclear Fuel

Besides the Area-25 facilities, DOE evaluated an elaborate spent nuclear fuel handling system as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would be located in Test Area 5, near the eastern border of the site, and in the general proximity of the low-level waste/low-level mixed waste disposal areas. The spent nuclear fuel complex would include:

- Spent Nuclear Fuel Receiving and Canning Facility,
- Technology Development Facility,
- Interim Dry Storage Area, and
- Expended Core Facility, similar to the one at Idaho National Engineering Laboratory.

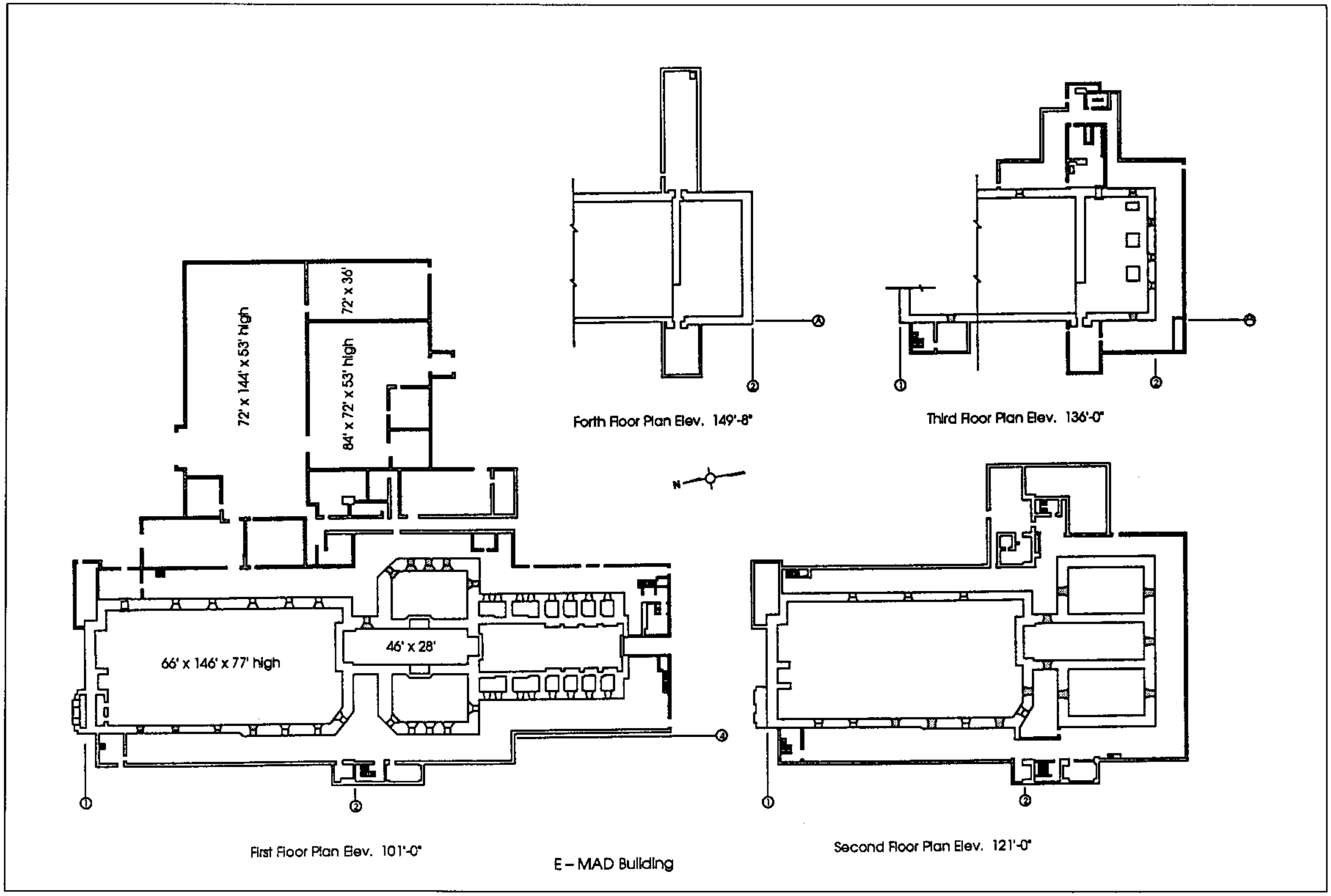


Figure F-24 Schematic of the Engine Maintenance and Disassembly Facility at the Nevada Test Site

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not reported.

Foreign research reactor spent nuclear fuel represents less than 1 percent of the DOE spent nuclear fuel quantities in terms of mass (i.e., potential source term), and about 10 percent in terms of volume. Thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated.

Figure F-25 summarizes the Nevada Test Site storage capabilities for foreign research reactor spent nuclear fuel. The Area-25 facilities could all receive and provide for dry storage of foreign research reactor spent nuclear fuel close to the proposed Yucca Mountain repository. It should be noted that these facilities have comparable shielded floor areas and volumes as compared to the generic modular dry vault

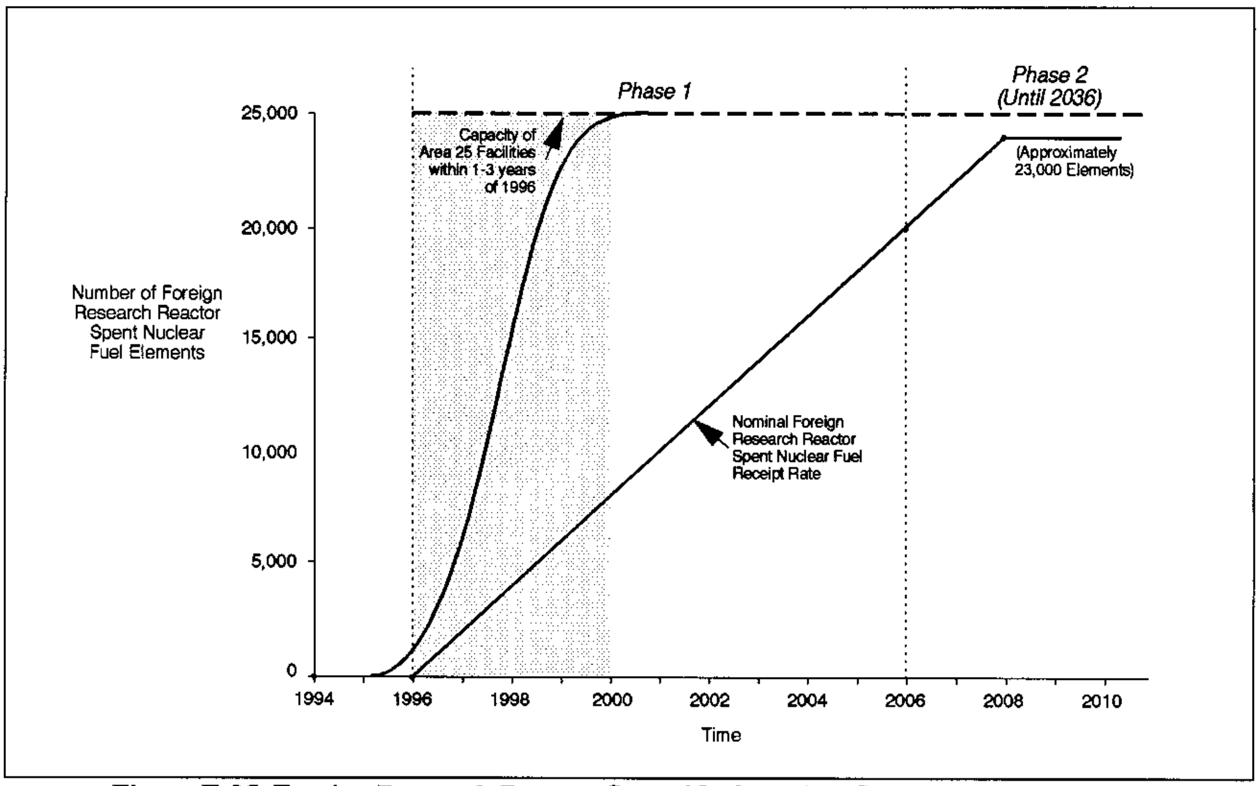


Figure F-25 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Nevada Test Site

for foreign research reactor spent nuclear fuel discussed in Section F.3. Alternatively, new facilities could be built, but these would require a longer transportation path to the proposed Yucca Mountain repository.

F.1.3.6 Storage at Overseas Facilities

Currently, foreign research reactor spent nuclear fuel is being stored in wet pools at foreign research reactor sites. These pools are approaching the levels of their capacity, which is why the foreign research reactor operators would like the United States to accept their spent nuclear fuel. An alternative being considered by DOE is foreign research reactor spent nuclear fuel storage at overseas facilities. Several facilities exist in Europe for contractual storage of both commercial and research reactor spent nuclear fuel for a fee, including:

- British facilities at Dounreay, Scotland and Sellafield, England. The former has several small pools for research reactor fuels, while the latter has several large pools with a capacity of 3,000 MTHM for commercial spent nuclear fuel (Bonser, 1994).

- French facilities at La Hague, with several large pools having a total capacity of 14,000 MTHM for commercial spent nuclear fuel (Nuclear Fuel, 1993); facilities at Marcoule, for research and metallic spent nuclear fuel.

Electricite De France has also announced its intention of constructing a commercial spent nuclear fuel wet storage facility with a capacity of 12,000 MTHM (Nuclear Fuel, 1994b). Dry storage of spent nuclear fuel is also being considered.

These facilities are predominantly stainless-steel lined wet storage pools that meet modern seismicity and confinement standards and maintain good water chemistry. Wet storage pools designed for commercial spent nuclear fuel could, after license modification and new rack installation, store foreign research reactor spent nuclear fuel. These overseas wet storage pools are similar in design and layout to the generic wet storage facility discussed in Section F.3.

F.1.4 Vitrified Waste Storage Facilities

If foreign research reactor spent nuclear fuel is processed, the resulting high-level waste would be vitrified and placed into stainless steel canisters. The Savannah River Site is the only domestic site that currently has a storage facility designed and built for storing vitrified high-level waste from the processing of spent nuclear fuel. This facility is termed the Glass Waste Storage Building, and it is located immediately adjacent to the Savannah River Site vitrification facility (the Defense Waste Processing Facility), in the S-Area of the site near the H-Area processing facilities (DOE, 1994g). Figure F-26 provides a general overview of the facilities in the S-Area. Figure F-27 displays a general layout of the building. The Glass Waste Storage Building is designed to accommodate the standard Defense Waste Processing Facility vitrified waste canister (Figure F-28). The existing building has space for 2,286 of these canisters. A second, almost identical building, is planned for construction starting in 2007. Additional buildings may be built, up to a total interim storage capacity of 10,000 canisters if delays in the Federal Repository Program are encountered (DOE, 1994g). The Defense Waste Processing Facility/Glass Waste Storage Building area does not currently include a cask receiving/shipping facility, but one is planned for future construction.

The facility is relatively simple in design and operation. It consists of a structure enclosing a concrete floor that functions as the charging face to the vault beneath it. Shield plugs are removed from the floor to provide access to storage tubes in the vaults that would contain the canisters. Each storage tube contains

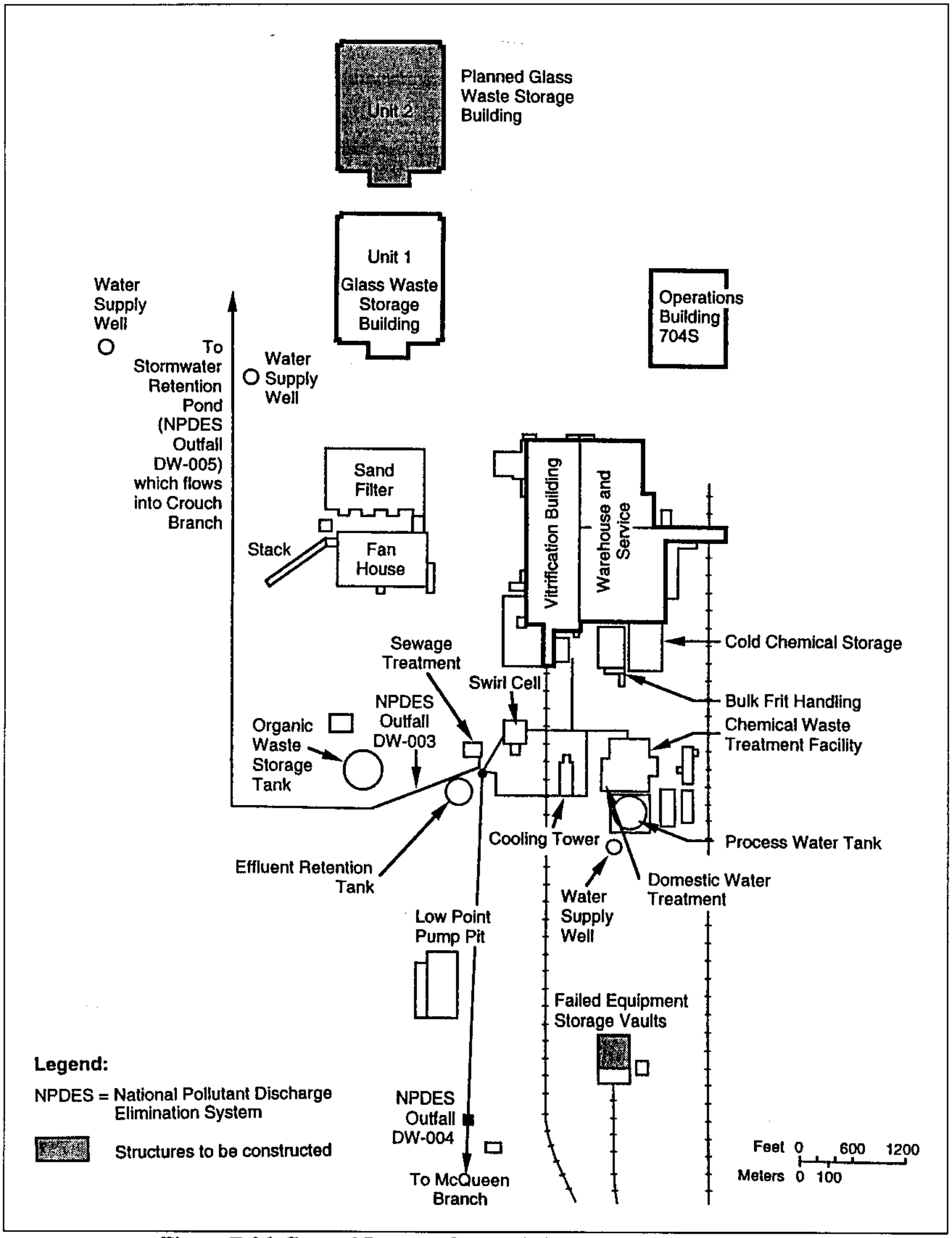


Figure F-26 General Layout of the Existing Vitrification Facilities at the Savannah River Site

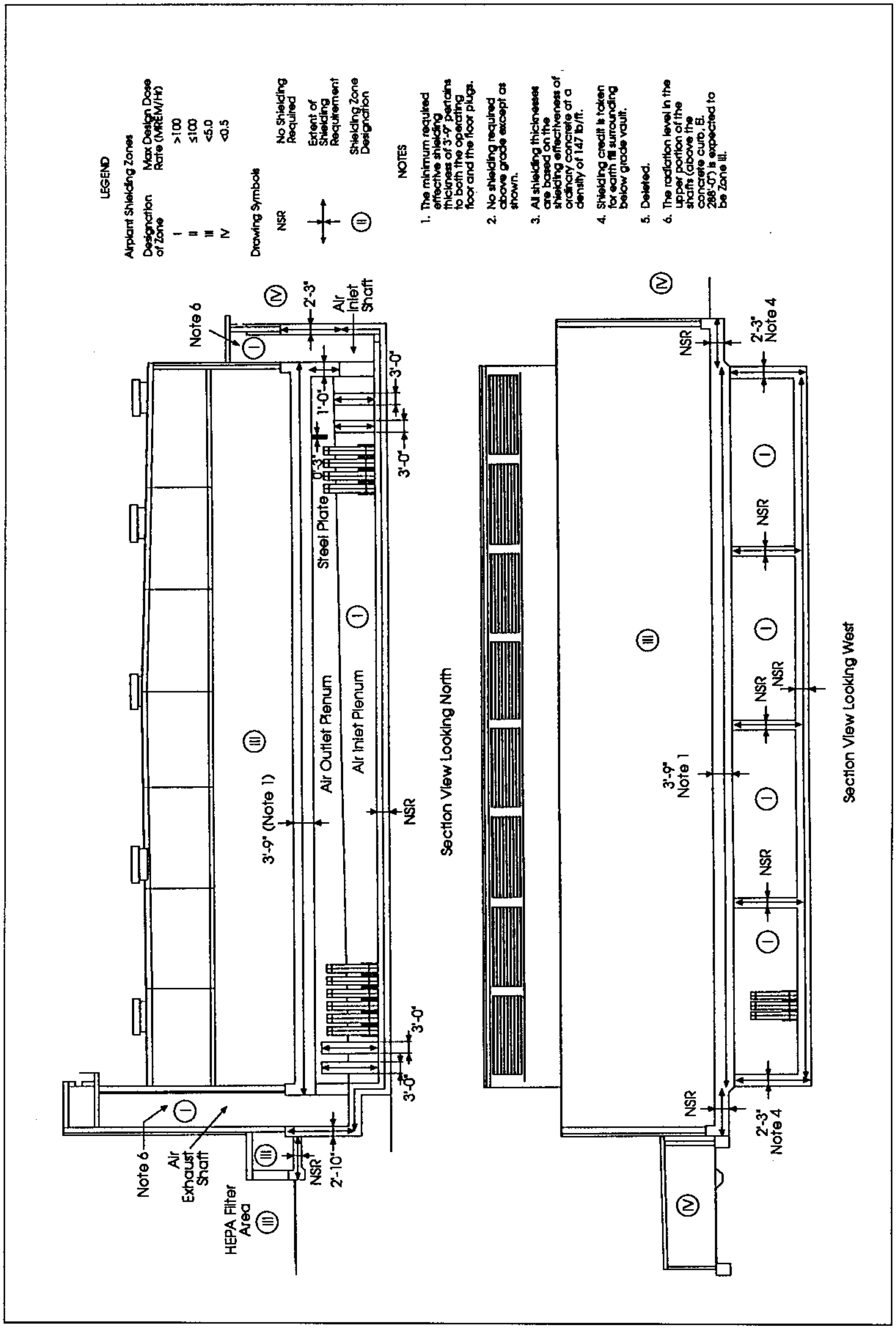
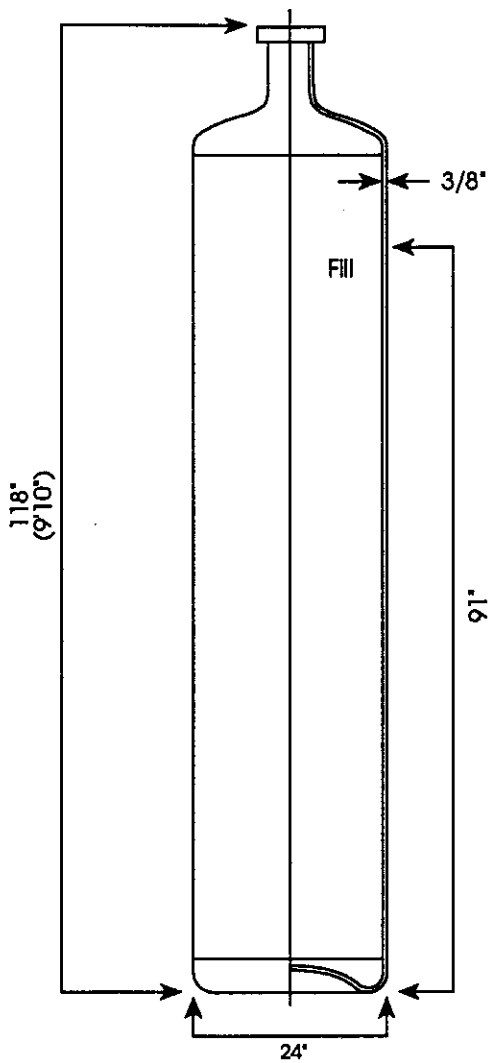


Figure F-27 Layout of the Glass Waste Storage Building



Material:	304L Stainless Steel
Empty Weight:	1,100 lb
Net Weight:	3,540 lb (Glass)
Weight of Radionuclide:	41 kg (90 lb)
Activity:	8.3×10^{15} Bq (223,000 Ci)
Decay Heat:	660 W
Radiation Field (at Surface):	6,300 rad/hr
Surface Contamination:	Less than 10^{-4} μCi/cm^2

Figure F-28 Typical Defense Waste Processing Facility Glass Waste Canister

two canisters, stacked vertically. The vault area consists primarily of steel-reinforced concrete and is designed to resist all earthquake and severe weather incidents.

Radioactive decay heat from the canisters is removed by the Glass Waste Storage Building's forced air fan exhaust system. The exhaust air is drawn around the canisters and then exhausted through the building's High Efficiency Particulate Air filtered ventilation system and discharged to the atmosphere via a stack. No condensate is expected to form, although the building does include a sump for exhaust air condensate. No radioactivity is expected in the exhaust air or in any condensates that might form.

During operation, a special dedicated transporter vehicle moves the canister from the Defense Waste Processing Facility vitrification building to the Glass Waste Storage Building in a shielded transporter. The transporter's cask is placed over the appropriate vault borehole, the shield plug is removed, and the canister is lowered via a crane mechanism into the borehole. The shield plug is replaced, and the transporter returns to the vitrification plant for the next shipment.

Several overseas facilities also exist for vitrified waste storage at Marcoule (France), La Hague (France), and Sellafield (England) (COGEMA, 1994a and 1994b; BNFL, 1994a and 1994b). These facilities are designed as natural circulation vaults and, thus, do not require fans for storage cooling. These vaults use a smaller canister, with several thousand currently in storage.

F.2 Storage Technology Evaluation Methodology

The selection of a spent nuclear fuel storage technology for foreign research reactor spent nuclear fuel requires a multi-disciplinary approach including the evaluation of, at the minimum, the environmental impact of alternatives and the following key design and performance areas:

- chemical compatibility,
- subcriticality assurance,
- shielding effectiveness,
- structural integrity (i.e., containment),
- thermal performance,
- ease of use,
- cost, and
- regulatory basis and licensing.

Other factors that may affect the decision process are whether the design has been previously licensed and actually used to store spent nuclear fuel, and its perceived ability to meet applicable regulations and standards if it has not yet been licensed.

Two principal types of spent nuclear fuel storage can be used for foreign research reactor spent nuclear fuel, wet and dry. Wet storage denotes the immersion of fuel in a pool of water, which performs the dual functions of shielding/leaking radionuclide removal and decay heat removal, but which relies on active systems. Dry storage encompasses a wide spectrum of structures that house the fuel in a dry inert gas environment, with an emphasis on passive system design and operation.

F.2.1 Chemical Compatibility

The most important criterion in assessing foreign research reactor spent nuclear fuel storage technologies is the compatibility of the spent nuclear fuel with the fuel storage technology environment. The research reactor fuel cladding is either aluminum or stainless steel. Aluminum cladding fuel is the predominant type in the mix of foreign research reactor spent nuclear fuel being considered for acceptance in this EIS. The selected method of storage for any foreign research reactor spent nuclear fuel that may be accepted must provide a benign and noncorrosive environment for the fuel.

In reviewing the corrosive potential of aluminum, acidic, alkaline, and even many neutral chemical solutions have been found to be significantly corrosive. Therefore, the use of wet storage technology for the majority of the foreign research reactor spent nuclear fuel that contains aluminum cladding would require the maintenance of high water purity throughout the storage life of the pool, which may equal or exceed 40 years.

Unlike wet storage, most of the dry storage technologies utilize a dry inert gas atmosphere for the fuel, which is a noncorrosive noble gas that also enhances conduction heat transfer from the fuel to the encapsulating container. Some dry storage technologies use dry nitrogen instead of inert gas. Even in the event of a loss of inert gas atmosphere, the air atmosphere would be less corrosive than a less-than-high-purity water pool. Finally, previous experience at many DOE wet storage facilities has shown that poor water quality dramatically deteriorates the integrity of aluminum fuel. Thus, the chemical compatibility criterion indicates that dry storage is a more appropriate technology than wet storage for extended storage of foreign research reactor spent nuclear fuel.

F.2.2 Subcriticality Assurance

Uranium and plutonium are the principal elements that have the unique ability to split or fission after absorbing a neutron, and release energy and several new neutrons from this fission process. The particular forms or isotopes of uranium that are effective in the fission process are called fissile materials and are ^{233}U, ^{235}U, and ^{239}Pu. Of these three isotopes, only ^{235}U exists naturally, while the other two isotopes can be produced artificially. Under the right conditions, the fission process can be self-sustaining or even grow by a chain reaction. This chain reaction produces as many or more neutrons than are absorbed in an assembly of fissile materials.

In nuclear engineering terminology, the numerical measure of a mass of fissile material to achieve and maintain a self-sustaining fission chain reaction is termed K-effective. K-effective is the net ratio of neutrons produced per neutron absorbed in the fissile material mass. When K-effective equals 1.0, the mass is said to be critical because it can maintain the fission process. When K-effective is less than 1.0, the mass is considered to be subcritical.

Subcriticality can be ensured by a number of factors, including:

- diluted concentration of fissile materials,
- adequate separation distance between masses of fissile materials such as nuclear fuel rods or assemblies,
- presence of materials (such as boron) mixed with the fissile material that absorbs neutrons before they can be captured by the fissile material,

- exclusion of substances such as water that can encourage the absorption of neutrons by the fissile isotopes or reflect neutrons leaving the mass of fissile materials back into the mass, and
- restricting the mass of fissile material below the minimum that nature requires to initiate, maintain, and/or sustain a fission chain reaction.

In nuclear criticality safety, the principle of double contingency is used to protect against criticality. Double contingency requires that the design of any system containing fissile material use two of the aforementioned factors to prevent the onset of criticality. Criticality analyses would be required to confirm spacing and the effects of optimum moderation, as well as the different structural materials in foreign research reactor spent nuclear fuel as compared to commercial fuel (e.g., aluminum, stainless steel, hydride, inconel in foreign research reactor spent nuclear fuel as compared to zircaloy in commercial fuel).

Another important factor is that most of the products of the fission reaction are radioactive fission products that are not capable of sustaining a fission reaction. Like most elements in nature, these fission products can absorb neutrons, but do not produce any neutrons or energy during this absorption. Mixed with these fission products is a small amount of fissile ^{239}Pu, which was also created during the fission process. However, the sum of the remaining ^{235}U and the created ^{239}Pu is much smaller than the original quantity and concentration of ^{235}U in the new fuel. The relatively low (in comparison to its initial value) ^{235}U enrichment and ^{239}Pu present in foreign research reactor spent nuclear fuel, coupled with the presence of neutron-absorbing fission products, greatly reduces the physical ability of foreign research reactor spent nuclear fuel to become critical.

Another important aspect in selecting an appropriate storage technology is the maintenance of subcriticality. For wet (pool) storage, subcriticality is ensured by fuel spacing, and in some cases, the use of spacing plates between adjacent fuel assemblies that contain boron. In addition, control of the maximum allowable concentration of fissile isotopes (i.e., ^{235}U enrichment) is another method used to control subcriticality. Since all dry storage technologies use a storage canister for fuel, the subcriticality design relies on controlling the fissile material inventory, fuel spacing, and if necessary, the use of neutron-absorbing materials. The subcriticality control design of all fuel storage technologies is acceptable and does not provide any discriminating factors for selecting one technology over another.

F.2.3 Shielding Effectiveness

Shielding effectiveness design impacts both onsite worker and public dose rates during the loading and subsequent storage of spent nuclear fuel. Both neutron and gamma ray shielding must be provided and ensured throughout the life of the storage facility. Wet storage technology uses pool water as a shield, which effectively reduces both neutron and gamma doses to acceptable levels. The only weakness of this shielding design is any event in which the water could be lost due to a leak in the pool wall or, if the design includes piping at a low enough level, a pipe break in a line connected to the pool for the water purification or decay heat removal systems. Precluding piping or wall leakage, spent nuclear fuel pool water is an inexpensive shield medium that offers the advantage of providing visual inspection of the stored fuel as long as water purity and clarity are maintained.

Dry storage technology relies on a number of solid shielding materials, sometimes in combination, to reduce gamma and neutron dose rates. The most common materials are different forms of concrete (i.e., low-density, high-density, hydrogenated), cast iron, carbon or stainless steel, lead, borated resin, and polyethylene (for neutrons). As with water, these materials have been widely used in the nuclear industry for shielding and their properties are well known. They are more costly than pool water and prevent visual inspection of the spent nuclear fuel, but are not prone to material loss like pool water.

In comparing shielding designs, it is important to note that most of the shielding materials have inherent limiting temperatures (i.e., maximum allowable temperature) with the exception of the steels and cast iron. These metals' temperature limits are much greater than the aluminum-based fuel cladding temperature limit. Shielding material thermal limits include both absolute values of temperature and, in the case of concrete, temperature gradients that create thermal stresses. Wet storage pool water also has a thermal limit that is the prevention of local or bulk boiling in the pool. Operation of the spent nuclear fuel pool heat removal system prevents pool water boiling, but a postulated accident in which this system is disabled requires calculation of the time before the inception of bulk pool boiling. Adequate natural convection between adjacent fuel assemblies and within storage racks prevents local nucleate boiling in any fuel flow channel.

Shielding geometry plays an important role in the determination of a dose rate profile around the storage facility. A continuous and constant thickness of shielding completely surrounding the fuel provides a relatively constant dose rate at all locations. A shielding design that is asymmetric and contains air gaps and/or varying material thickness results in hot spots and a relatively larger variation in surface dose rates. The wet pool design offers a continuous shield of water with resulting low constant dose rates throughout the pool surface. The water and pool wall, usually steel-lined concrete, also maintain a low continuous dose rate profile outside the walls.

The dry storage concrete building and concrete cask technologies rely on concrete walls for shielding with some steel internal to the walls. The need for internal airflow passages in the concrete introduces gaps in these walls. These gaps, which are labyrinths, require complex shielding analyses and typically allow a relatively larger dose rate at the air inlets and/or outlets than at the bulk concrete wall. This effect is more significant for concrete casks than for concrete buildings because the casks are more limited in the concrete thickness that is used in their shield wall. In the case of metal casks, since there are no internal air passages in the metal shield, the dose rate is relatively uniform around the surface. Different axial shielding and neutron-gamma source terms will result in different axial dose rates for the metal and concrete casks. Inground storage systems use a relatively small amount of concrete radially coupled with the surrounding earth for shielding and employ thick steel plugs for axial shielding. The hybrid metal-concrete cask design uses shielding principles similar to the concrete cask.

In comparing spent nuclear fuel storage shielding designs, the four basic technologies can be characterized as water, lead, metal, and concrete. Water and metal provide the most uniform dose rate reduction because they do not require the inclusion of labyrinth airflow passages for decay heat removal necessary for concrete. Water is most susceptible to a sudden rapid loss of shielding effectiveness because it is a liquid requiring confinement. It should be noted, however, that pool storage of spent nuclear fuel has been effectively used in the nuclear industry for over 40 years. Both concrete and water are susceptible to degradation of their shielding effectiveness if temperature limits are exceeded. These thermal limits can be accommodated by proper design of the spent nuclear fuel pool cooling system for water shields and conservative design along with airflow passage surveillance for the concrete shield. The shielding properties of all three shields are well known and therefore not subject to significant design uncertainties.

Shielding design is dictated by the regulatory dose limits, maximum bounding radiological neutron and gamma ray source terms of the fuel to be stored, cost, weight (in some cases), and thermal limits of some shielding materials. In general, the commercial nuclear power spent nuclear fuel storage technologies discussed in this appendix were designed to provide adequate shielding for fuel assemblies containing several hundred thousand curies (Ci) of fission products per assembly. The foreign research reactor spent nuclear fuel being considered for acceptance in the United States will contain fission product inventories of from 1,000 to 100,000 (maximum) Ci per assembly. Therefore, the radiation source term for shielding design purposes, assuming the same number of fuel assemblies in each storage technology unit, may be significantly smaller for foreign research reactor spent nuclear fuel than for commercial fuel. The cost savings associated with a reduction in shielding thickness are expected to be more significant for the metal cask and concrete building designs because of their relatively higher costs. At the present time, it appears appropriate to use available designs from the vendors.

Based on the aforementioned vulnerabilities, the best shield would be the metal cask. Concrete shields are judged second best after metal based on their lack of dependence on any active systems. The water shield requires active systems for decay heat removal to prevent heatup and makeup to compensate for long term evaporation. It is also vulnerable to leaks from connected piping and its enclosing structure. Although water appears to be the least expensive shield material, its requirements for several active systems and qualified walls and floor actually make it one of the more expensive shields.

F.2.4 Structural Integrity

All of the spent nuclear fuel storage technologies are required to meet the same standards for structural integrity in accordance with appropriate codes. Structural integrity ensures that the confinement boundary around the spent nuclear fuel is maintained under all operational and accident conditions.

For incident-free operation, the dry storage designs are analyzed in terms of peak stresses on their canister and enclosing structure (i.e., metal cask, concrete cask, or vault). In wet storage designs, the fuel racks and pool structure are analyzed for operating loads. The source of these loads, in accordance with appropriate American Society of Mechanical Engineers codes, include such factors as deadweight, pressure, fill gas pressure, and thermal gradients.

For accident cases, additional loads are imposed upon the structures. These additional loads include seismic acceleration, high (or low) ambient temperature and solar heat flux, component drop or tip over, airflow passage blockage, external fire, tornado missile, flooding, etc. As with incident-free operation, specific prescribed margins of safety between the peak calculated stresses and the maximum allowable stress for a given component, location, and material must be maintained to substantiate structural integrity.

The principal structural-related differences between foreign research reactor spent nuclear fuel and commercial fuel for storage technology design purposes are:

- a typical foreign research reactor spent nuclear fuel element is much lighter [5 kg (11 lbs) as compared to 800 kg (1,760 lbs)] and shorter than a commercial fuel assembly (a stack of 5 typical foreign research reactor spent nuclear fuel elements is approximately equal in length to 1 commercial fuel assembly), and

- the strength of foreign research reactor spent nuclear fuel, in particular the predominant aluminum-clad design, is expected to be less than the commercial fuel assembly.

The much lower foreign research reactor spent nuclear fuel weight will reduce the total weight and load on the storage technology unit by about 19 metric tons (21 tons) for a 24-commercial fuel assembly design. For metal and concrete casks, this is a significant fraction of the total cask's weight, and can only improve the structural strength of the cask. The lower weight of the foreign research reactor spent nuclear fuel will increase the structural margins in the design and possibly allow for the use of less material in the structure compared with the commercial cask design. Any design changes to take advantage of the lower fuel weight would require detailed re-analysis, and are probably unnecessary.

The lower strength of the foreign research reactor spent nuclear fuel would require analyses to demonstrate that operational and postulated accident events do not result in structural failure of the fuel. However, since the principal means of confinement is the canister surrounding the fuel, its structural integrity is expected to be maintained, as it has already been qualified for the heavier commercial fuel under the same conditions and accidents.

Assuming that the same structural design limits apply for foreign research reactor spent nuclear fuel storage as for commercial fuel storage, the lower weight and strength of the foreign research reactor spent nuclear fuel would be expected to increase the original stress design margins.

The basket of any currently licensed cask would require redesign to accommodate the foreign research reactor spent nuclear fuel. Furthermore, it could be anticipated that permanently installed neutron poisons may be required in the basket to prevent criticality for the highly enriched fuels (initially 90 to 93 percent enrichment).

Each of the spent nuclear fuel storage technology designs that have been licensed by the NRC have undergone rigorous structural analyses and have been shown to meet all applicable standards and codes. Designs which have not yet been licensed would be required to present detailed structural analyses for review and confirmation to ensure structural integrity. No design has specific structural vulnerabilities that make it unsuitable for the storage of foreign research reactor spent nuclear fuel. It should be noted that any changes in existing NRC-approved storage designs that are deemed to impact stresses (i.e., reducing shielding wall thickness) would require extensive re-analysis and technical review for structural integrity. Thus, use of existing designs is favored.

F.2.5 Thermal Performance

Adequate decay heat removal is vital to preventing degradation of the fuel cladding barrier to fission product releases. The wet and dry storage technologies rely on a combination of conduction, convection (natural or forced), and radiation heat transfer mechanisms to ensure fuel cladding temperatures below appropriate long term storage limits.

In wet pool designs, fuel decay heat is transferred to the pool water by conduction and natural convection, which is induced by the axial enthalpy rise of the water as it passes over the active region of the fuel. An active cooling system consisting of redundant pumps, heat exchangers, and piping connected to the pool removes the heat in the bulk pool water. Careful thermal design of the spent nuclear fuel storage racks allows for sufficient natural convection flow over each fuel assembly to prevent any local nucleate boiling on the cladding surface throughout the pool. Therefore, the thermal performance of the pool technology relies on storage rack design for local thermal effects and an active external system for global heat removal. As previously discussed, this design has a long-established history of satisfactory performance. Wet storage can accommodate fuel of any power level.

The metal cask, dry storage design relies on a totally passive system for heat removal. The fuel decay heat, in an encapsulating inert gas atmosphere canister, is transferred to the canister's walls by a combination of radiation and conduction heat transfer. The canister walls, in contact with the metal (or sometimes metal sandwiched with a neutron-absorbing material) cask wall transfers this heat by conduction through the metal wall. At the outside of the metal cask, the heat is removed by conduction and natural convection to the environment. Some designs incorporate cooling metal fins on the exterior of the cask to enhance heat transmission to the air. The four metals used in spent nuclear fuel storage cask designs are ductile cast iron, carbon steel, lead, and stainless steel. In terms of their heat conduction properties, cast iron, lead, and carbon steel are superior to stainless steel because they have a thermal conductivity which is about three times that of stainless steel. The metal cask heat transfer system is not susceptible to thermal limits, since these metals have a higher temperature limit than fuel cladding. The only possible degradation of heat transfer could occur if the fuel canister seal was broken and the inert gas atmosphere lost. The sealing system is designed to withstand all postulated accidents and maintain integrity over the lifetime of the cask, because it constitutes part of the radioisotope confinement boundary.

As with metal casks, concrete casks use a passive heat removal system, but the concrete cask system has one inherent vulnerability. To remove fuel decay heat and stay below both the fuel cladding and concrete temperature limits, concrete casks must include a labyrinth airflow passage design that allows natural convection-driven air to enter the cavity enclosing the canister inside the concrete. The air then exits through higher elevation paths through the concrete to the environment. Concrete thermal conductivity is a factor of 10 to 40 lower than that of the previously discussed cask metals. The need for these airflow passages and their associated inlets and outlets introduces the possibility of an accident in which the inlet and/or outlets could be blocked by debris, snow, or even nests or hives. Therefore, concrete casks require surveillance of their air inlet and outlet flow passages. Typically, measurement of the air temperature rise between the inlet and outlet is also used to validate the thermal design and as an operating specification. The elevation difference between the air inlets and outlets is an important design factor in the effectiveness of natural convection-driven airflow through the cask. Larger elevation differences induce a greater airflow rate, which improves heat removal. The above discussion would not apply to a solid concrete design (such as a SILO), because it does not use internal airflow passages for decay heat removal; it uses only concrete conduction.

In a concrete cask, heat transfer within the canister is identical to that of the metal cask. The canister transfers heat by conduction, natural convection, and radiation heat transfer to the airflow around it and the concrete walls surrounding it. That portion of the heat transferred to the concrete walls is then conducted through the concrete to the outside air. The concrete building technology uses a heat transfer system identical to that of the concrete cask. However, its relatively larger size, translating to a larger flow area for air inlets and outlets, makes it less susceptible to flow passage inlet and/or outlet blockage. Also, the concrete building designs typically incorporate a much larger elevation difference between air inlets and outlets than the concrete cask, which further enhances natural convection flow driven heat transfer.

For commercial nuclear fuel, the long-term storage temperature limits are well known (Levy et al., 1987; Johnson and Gilbert, 1983; Einziger and Cook, 1985; and Kohli et al., 1985), and typically about 350^{o}C (662^{o}F). The shielding material limits usually apply to concrete or concrete-like materials, but may also apply to resins or polyethylene. Shielding material temperature limits are not affected by the use of foreign research reactor spent nuclear fuel instead of commercial nuclear fuel in the storage technology. However, the thermal limits of the TRIGA and MTR foreign research reactor spent nuclear fuel could affect the thermal design.

Currently, there is limited well-documented information available on long-term foreign research reactor spent nuclear fuel storage temperature limits for fuel cladding. However, the aluminum cladding of the MTR-type foreign research reactor spent nuclear fuel has a much lower melting point than commercial nuclear fuel zircaloy cladding [649^{o}C (~1,200^{o}F) for aluminum versus 1,832^{o}C (~3,330^{o}F) for zircaloy]. Thus, the maximum long-term storage temperature limit for aluminum-clad foreign research reactor spent nuclear fuel would be expected to be considerably lower than that for zircaloy-clad commercial nuclear power fuel. Aluminum also undergoes a phase change at around 250^{o}C (482^{o}F), which results in a reduction of its tensile properties. An offsetting physical property of aluminum that may partially compensate for its lower melting point is that aluminum has a thermal conductivity more than 10 times greater than zircaloy. This would tend to reduce the temperature difference across the aluminum cladding as compared to zircaloy cladding. TRIGA foreign research reactor spent nuclear fuel cladding is composed of stainless steel or inconel, which have similar thermal conductivities to zircaloy, but a melting temperature of about 1,371°C (2,500°F). TRIGA fuel storage temperature limits are expected to be greater than for aluminum-clad fuel. The Savannah River Site is conducting a research and development project, initiated in FY 1994, to examine the applicability of aluminum-clad spent nuclear fuel dry storage.

At a minimum, a new thermal analysis would need to be performed for existing designs of spent nuclear fuel storage technologies. This analysis would use the parameters associated with foreign research reactor spent nuclear fuel instead of commercial spent nuclear fuel. The important changes in thermal performance parameters for the foreign research reactor spent nuclear fuel are:

- lower individual fuel assembly decay heat power,
- lower and/or different fuel temperature limits for aluminum-clad and stainless steel-inconel-clad fuels, and
- higher clad and fuel thermal conductivity for aluminum foreign research reactor spent nuclear fuel.

A temperature limit of 175°C (347°F) has been tentatively identified to avoid damage to the cladding of aluminum-clad spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). The results of this revised thermal analysis could impact the thermal design of the spent nuclear fuel storage technology. If the existing design results in unacceptable fuel and/or shielding temperatures, redesign could reduce the maximum heat load of each module or cask or increase the airflow passage area or height for concrete casks that rely on natural convection heat transfer. The new thermal analysis must take into account design restrictions that are imposed by criticality limits (i.e., the maximum allowable number of foreign research reactor spent nuclear fuel assemblies), and possible changes in shielding thickness due to lower gamma and neutron source terms that would improve the storage technology's thermal performance. Again, existing designs should be used to the greatest extent possible.

Commercial spent nuclear fuel dry storage systems require a minimum cooldown period of 5 years. For aluminum-clad foreign research reactor spent nuclear fuel, the preliminary cladding temperature limit of 175°C (347°F) becomes the determining criteria for dry storage loading above an average spent nuclear fuel element power level of 40 Watts each. Foreign research reactor spent nuclear fuel averages more than 40 Watts per element after a single year's discharge from the reactor and, if immediately placed into dry storage, would result in oversized facilities within several years as the radionuclides decay. Consequently, for the size of a foreign research reactor spent nuclear fuel dry storage facility to be minimized, an average foreign research reactor spent nuclear fuel power level below 40 Watts per element is necessary. On average, a 3-year cooldown period would be required. This results in the element's volume being the constraining criteria, and corresponds to maximum density of spent nuclear fuel (hence, minimum size of the facility) in the dry storage method. Consequently, the storage approach uses a minimum wet storage period of 3 years prior to emplacement into dry storage.

A comparative evaluation of the thermal performance of each fuel storage technology points to the metal cask and the solid concrete SILO as the simplest, effective, and least susceptible to any degradation. However, another design which has many merits is the concrete building. Although concrete buildings require open airflow passages to remove decay heat, size and a large elevation difference are factors which compensate for this weakness and make them good candidates. The concrete cask, with adequate design margins and surveillance is an acceptable thermal system. Finally, the wet pool system is a proven technology, but is dependent on an active system to remove heat. The inground concrete system in

Denmark (RISO National Laboratory) relies on a forced air active system and is characterized similar to the wet system in terms of its heat removal capabilities.

F.2.6 Ease of Use

For spent nuclear fuel storage, ease of use is defined as the lack of complexity involved in the process of loading spent nuclear fuel, and operating and maintaining the storage technology. For all storage designs, the spent nuclear fuel must be removed from the transportation cask to be placed into the storage facility, unless the design is a dual-purpose cask.

The technology that requires the fewest steps and lowest complexity for transferring spent nuclear fuel from the transportation cask to its storage location is wet pool storage. At a pool, the transportation cask is simply immersed under the water, opened underwater, and the fuel moved underwater to its final location in a storage rack in the pool. Pool water provides shielding, heat removal, and viewing of the fuel. The dry storage technologies all require additional intermediate steps, which include the insertion of fuel into a canister that must be subsequently drained of all water and air, seal welded, tested for leakage, and backfilled with inert gas. The canister is then placed into its dry storage structure (i.e., vault, concrete, or metal cask). The vault provides for this entire process within a shielded enclosing building, whereas the casks require transport by some vehicle between the transportation cask fuel transfer location and the cask site. Thus, for spent nuclear fuel transfer and loading, the wet storage design is easiest to use, followed by the dry vault.

After loading, operation of the storage facility is another important factor in determining ease of use. For operation, the individual metal or concrete casks are easiest, since they are designed as totally passive systems requiring only periodic visual inspection from a distance. The vault is slightly more complex than the casks because it includes a number of active systems (i.e., crane, power supplies, fuel handling machine) that may require some operational support. The wet storage is the most complex from an operational viewpoint because it includes a number of vital safety-related systems that must be monitored and controlled (e.g., heat removal system, water purification system, makeup water system, ventilation exhaust system).

Maintenance ease of use is closely related to operational ease of use since designs with more operational complexity require greater maintenance. Thus, the cask systems can be considered easiest, followed by the vault, and the wet technology.

In ranking the relative importance of the three aforementioned factors of ease of use (fuel loading, operation, and maintenance) the fraction of time spent during the life of the facility for fuel loading is expected to be much smaller than for operation and maintenance. Fuel loading will be a sporadic event over a long period of time, whereas operation and maintenance are considered continuous over this same period of time. Thus, operation and maintenance ease of use is given greater importance than fuel loading ease of use. With this ranking, the cask (both metal and concrete) technology is judged to have the greatest ease of use, followed by the vault system. The wet storage technology is judged to have the lowest ease of use principally because of its safety-related active systems.

F.2.7 Cost

Information on the cost of different spent nuclear fuel storage technologies is limited because of its proprietary nature, but several comparative statements apply to the different designs. Operation and maintenance costs are expected to be highest for those technologies that rely on active systems for safety. Thus, the wet pool and inground forced air technologies have higher operations and maintenance costs than dry metal casks, dry concrete casks, and dry concrete buildings. The dry concrete vault/building technology would be expected to have slightly higher operations and maintenance costs than the individual metal or concrete casks, since these buildings use active nonsafety systems such as lighting, cranes, and fuel drying dedicated to the vault facility.

For the construction of a new fuel storage facility for the purpose of storing foreign research reactor spent nuclear fuel on the order of approximately 23,000 assemblies, elements and/or rods, it is assumed that 5 "trimmed" foreign research reactor spent nuclear fuel assemblies would occupy the same approximate space as one commercial nuclear power plant fuel assembly (Boiling Water Reactor-type). "Trimmed" means that the non-essential portions (i.e., ends) of the spent nuclear fuel element have been removed, as detailed in Appendix B. Therefore, storage of the total amount of foreign research reactor spent nuclear fuel under consideration in this EIS would be the equivalent of about 5,000 commercial Boiling Water Reactor spent nuclear fuel assemblies. Since typical concrete, inground, or metal casks can store 52 power fuel assemblies, this foreign research reactor spent nuclear fuel inventory would require around 100 casks. A suitably sized single pool or concrete building could accommodate this inventory of spent nuclear fuel. Spent nuclear fuel storage manufacturers have indicated that metal casks typically cost about twice as much as concrete casks for the same quantity of fuel storage due to the higher costs of metal as compared to concrete. Based on its design, the least expensive concrete cask is expected to be the simple concrete SILO, since it does not have steel-lined internal air passages. The number of foreign research reactor spent nuclear fuel assemblies under consideration in this EIS may be amenable to the economic advantages that a single building or pool offers over a large number of individual casks. Another potential cost advantage of the pool or concrete building is that these are self-contained, not requiring access to any other facilities for the transfer of the fuel from the transport cask. Presented below is a brief summary of commercial cost experience with storage.

F.2.7.1 Costs for Dry Storage Designs

The cost for different spent nuclear fuel storage technologies varies significantly between designs. Some information on cost has been obtained from manufacturers and openly available literature. Relative order of magnitude cost data was obtained for the horizontal concrete NUHOMS module, vertical concrete Ventilated Storage Cask design, vertical concrete SILO, metal CASTOR vertical cask, and the modular dry vault concrete building design.

The principal elements of cost that should be considered for the storage of foreign research reactor spent nuclear fuel are: (1) engineering for redesign and licensing, (2) capital for the construction of the facility, and (3) operations and maintenance. In the interest of minimizing cost and schedule for the completion of any storage facility for foreign research reactor spent nuclear fuel, the licensing basis of 10 CFR 72 used by the NRC for commercial nuclear power plant spent nuclear fuel should be adopted for the foreign research reactor spent nuclear fuel. This regulation provides all the requirements for licensing foreign research reactor spent nuclear fuel storage and has been successfully applied to numerous dry spent nuclear fuel storage installations in the United States.

Redesign engineering should be limited to changes in the design of the basket that encapsulates the fuel, since foreign research reactor spent nuclear fuel has different dimensions, would probably be stacked, and could require different spacing and/or the incorporation of neutron absorbing plates to maintain subcriticality safety margins. Outside the basket, all remaining components should be identical to those already licensed for commercial nuclear fuel by the NRC, thereby significantly reducing engineering analysis and license review time and costs as well as drawing and specification changes. This could result in some overdesign in the shielding and heat removal of the system, but would have the benefit of greatly reduced engineering, licensing, and schedule costs. If thermal analyses show that unacceptable foreign research reactor spent nuclear fuel temperatures would occur in the storage facility, then more extensive redesign would be required (e.g., reduce excess concrete wall thickness not needed for shielding, which then improves the conduction heat transfer) or fewer fuel assemblies could be stored in each unit.

Information obtained on the unit capital costs for different storage designs shows a significant variation. The least expensive unit is the SILO due to its simple concrete-canister design and lack of internal air passage labyrinth. The most expensive unit cost, excluding the modular dry vault (which stores a larger number of fuel assemblies than the other storage designs), is the CASTOR metal cask, due to its use of a thick metal wall instead of concrete. The Ventilated Storage Cask and NUHOMS designs' costs fall between the SILO and CASTOR. If one were to rank, in decreasing order, the unit cost of the four cask designs, they would be: CASTOR, NUHOMS, Ventilated Storage Cask, and SILO. There is more than a factor of 10 difference between the SILO and the CASTOR.

An estimate of the capital costs for storing approximately 23,000 foreign research reactor spent nuclear fuel elements can be made with the following two assumptions. First, the average spent nuclear fuel assembly decay heat is between 10 and 40 Watts, which is reasonable and conservative based on the status of foreign research reactor spent nuclear fuel under consideration in this EIS. Second, five trimmed foreign research reactor spent nuclear fuel assemblies can be stacked to fit into the same approximate space as one commercial nuclear power plant spent nuclear fuel assembly (Boiling Water Reactor-type). Using these assumptions, 23,000 foreign research reactor spent nuclear fuel elements would require 375 SILOs, 100 VSC-24s, 100 NUHOMS-24Ps, or 150 CASTOR V21s. One sufficiently sized and designed modular dry vault would also accommodate the foreign research reactor spent nuclear fuel.

The operations and maintenance costs for all these designs are expected to be small based on utility experience in operating dry spent nuclear fuel storage at numerous sites throughout the United States. The passive nature of these designs eliminates the need for any control room or continuous monitoring. Once completed and loaded with fuel, the storage facility would not require any onsite staff. Remote security surveillance cameras, fences, and thermoluminescent dosimeters for radiation monitoring would be utilized. Some designs (Ventilated Storage Cask and NUHOMS) require a periodic visual inspection of the labyrinth airflow inlets and outlets to ensure that there is no blockage. Periodic security fence thermoluminescent dosimeter retrieval and analysis would also be an expected operating requirement. No onsite utility consumption would be necessary except for that used in security lighting and cameras. Under incident-free conditions, no significant maintenance costs would be anticipated for most of the designs, with the exception of the modular dry vault, where equipment used in the movement and encapsulation of fuel would require some periodic maintenance.

Table F-15 provides a summary of the dry storage costs. Full-Time Equivalent estimates assume full-time assignment, whereas the utility experience indicates only part-time assignment would be necessary. Thus, Table F-15 costs are extremely conservative.

F.2.7.2 Costs for Wet Storage Designs

Based on previous utility experience, the costs associated with the design, licensing, construction, operation, and maintenance of a new spent nuclear fuel pool are expected to be higher than most of the dry storage designs. This is due to the structural, equipment, and active system requirements of a pool that must also be enclosed in a properly qualified structure. The need for active operating safety-related systems at a wet facility increases the operations and maintenance costs. These important systems include water purification and chemistry, water heat removal, water level, and heating, ventilation, and air conditioning for the building. Table F-16 provides a summary of the wet storage costs. Thus, it can be stated that the overall costs associated with the selection of a new wet storage facility for foreign research

Table F-15 Summary of Dry Storage Facility Costs Based Upon Utility Experience[a]

	Approximate Unit Capital Cost Range, $	Approximate # of Canisters/Sleeves for Foreign Research Reactor Spent Nuclear Fuel	Total Capital Cost $M	Full-Time Equivalents, Loading/ Inspection[b]	Full-Time Equivalents Monitoring[c]	Other Annual Costs, $M	Total Annual Operating Cost $M[d,e]
Metal Cask	800,000-1.1M	150 (max)	165	15 (max)	3 (max)	1	3.7
Horizontal Dry Storage Cask	400,000-500,000	100 (max)	50	15 (max)	3 (max)	1	3.7
Vertical Concrete Storage Cask	350,000	100 (max)	35	15 (max)	3 (max)	1	3.7
Modular Dry Vault	13,000/tube	5 foreign research reactor/vaul t tube	65	15 (max)	3 (max)	1	3.7
SILO	100,000	375 (max)	37.5	15 (max)	3 (max)	1	3.7

Reference for costs: (EPRI, 1993)

[a] *Intermediate wet pool required for dry storage facility not included because utilities already possess an on-site pool*

[b] *One shift operation*

[c] *Monitoring based on one Full-Time Equivalent per shift*

[d] *Average Full-Time Equivalent cost of $150,000 per year*

[e] *Estimated absolute maximum from utility experience; around $1 million per year appears to be the average*

Table F-16 Summary of Wet Storage Costs

Approximate Facility Capital Cost	$80-100 million
Full-Time Equivalents Loading/Inspection	30
Full-Time Equivalents Operations/Monitoring	(in above)
Other Annual Costs	around $1million
Total Annual Operating Cost	$6-12 million

(Nuclear Fuel, 1994a)

reactor spent nuclear fuel could be significantly larger than for any comparably sized dry storage design using concrete.

F.2.8 Design, Construction, and Operational Requirements

The DOE has orders dealing indirectly with the storage of spent nuclear fuel. A search was also made to determine if other Federal requirements exist that specifically deal with spent nuclear fuel.

DOE Order 5400.1 (DOE, 1988), entitled "General Environmental Protection Program," establishes environmental protection program requirements. It applies to all Departmental elements and contractors performing work for DOE. Although the order makes no direct reference to spent nuclear fuel, Chapter IV deals with environmental monitoring requirements, which could be useful in establishing a legal framework for storing spent nuclear fuel.

DOE Order 5400.5 (DOE, 1990), entitled "Radiation Protection of the Public and the Environment," establishes standards and requirements for operations of the DOE and DOE contractors with respect to protection of the public from undue radiological risk. The provisions apply to all Departmental elements. This order references the storage of spent nuclear fuel. It also references instances where some DOE facilities are subject to provisions of 10 CFR 72. It was not made clear in the order which DOE facilities are subject to 10 CFR 72, which deals directly with all aspects of interim storage of spent nuclear fuel.

DOE Order 5480.22 (DOE, 1992b), entitled "Technical Safety Requirements," requires that DOE nuclear facilities delineate criteria, content, scope, documents, etc. The scope includes DOE elements, but excludes facilities exempt from NRC licensing and Naval Propulsion Program facilities. Although this order does not reference spent nuclear fuel, there are useful discussions of limiting conditions for operation of nonreactor nuclear facilities.

DOE Order 5633.3A (DOE, 1994e), entitled "Control and Accountability of Nuclear Materials," prescribes minimum requirements and procedures for control and accountability of nuclear materials at DOE facilities, which are exempt from NRC licensing requirements. By DOE definition, "nuclear materials" includes spent nuclear fuel. Storage of nuclear material is mentioned with respect to repositories.

DOE Order 6430.1A, (DOE, 1989a), entitled "General Design Criteria," has a section dealing with irradiated fissile material storage facilities (Section 1320). General criteria for nuclear criticality, confinement systems, effluent control and monitoring, and decontamination and decommissioning are discussed. Reference is made that, "the design professional shall consider the criteria provided in 10 CFR 72," (NRC, 1994) as well as NRC Regulatory Guides 3.49 (NRC, 1981) and 3.54 (NRC, 1984) for applicability to irradiated fissile material storage facilities. Other important standards for dry storage are ANSI/ANS-57.9 (ANSI, 1984a) and NRC Regulatory Guide 1.13 (NRC, 1975).

F.2.9 Aluminum-Clad Research Reactor Spent Nuclear Fuel Dry Storage Experience

F.2.9.1 Australia

Australia has successfully operated an underground dry storage facility for High Flux Australian Reactor MTR-type aluminum-clad research reactor spent nuclear fuel for 31 years at the Lucas Heights Facility (Australia, 1993; Ridal, 1994; Silver, 1993). The facility consists of a building enclosing a concrete floor with 50 steel plugs that are bolted to a steel collar set into the concrete.

Each plug covers a stainless steel-lined 0.64 cm (0.25 in) thick, 14 cm (5.5 in) inner diameter, 15.2 m (50 ft) deep borehole tube that is sealed at the bottom. The rock around these 50 borehole tubes is sandstone with a variable clay matrix and bands of enriched siderite. The actual boreholes in the sandstone are 16.5 cm (6.5 in) in diameter, 16.8 m (55 ft) deep, and spaced 1.14 m (45 in) center-to-center apart.

Each borehole liner is filled with 11 stainless steel canisters that hold 2 stacked fuel assemblies each. The borehole liner is evacuated and backfilled with dry nitrogen. The borehole liner plug is designed with its own plug to allow for atmosphere purging, backfilling, and annual monitoring of any fission product gases that would indicate canister breach.

The stored High Flux Australian Reactor spent nuclear fuel is uranium-aluminum alloy with aluminum cladding in the shape of four concentric tubes. Each fuel assembly has an outer diameter of 10 cm (3.93 in) and a length of 66 cm (26 in). The ^{235}U content for each fresh fuel assembly was 170 g (0.37 lbs), and the ^{235}U was enriched to 60 percent. Fuel has been stored at this facility for 8 to 31 years with no radioactivity releases or evidence of corrosion over this time period. No nuclear poisons for criticality safety or heat transfer analyses were deemed necessary because of the relatively low ^{235}U content, large borehole spacing, and low fuel assembly decay heat. The storage criteria for each fuel assembly is a maximum decay heat of less than 4.5 Watts, which after 20 years, drops to 1.5 Watts per fuel assembly. Fuel examined in a hot cell after 10 and 25 years of storage at this facility showed no visible signs of corrosion. Figure F-29 illustrates the facility design.

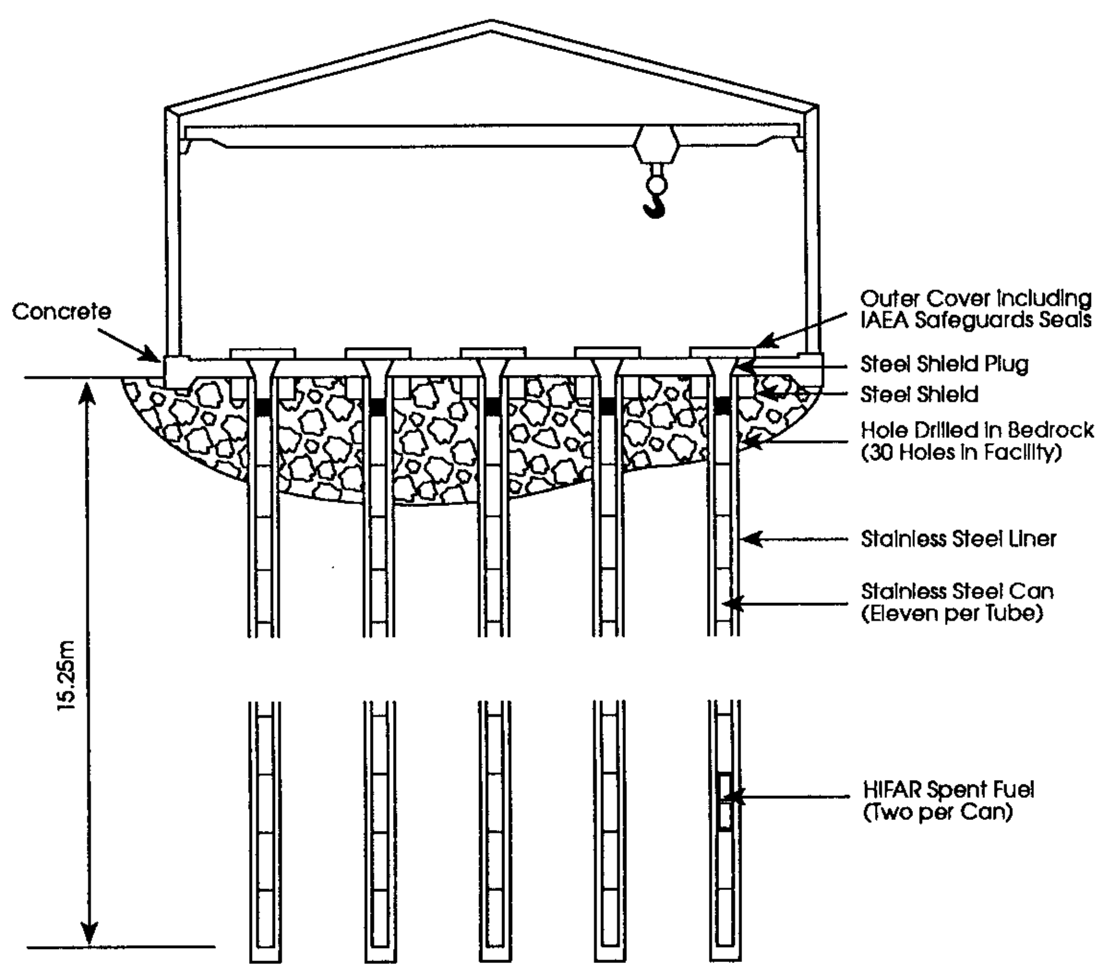


Figure F-29 High Flux Australian Reactor Spent Nuclear Fuel Dry Storage Facility

F.2.9.2 Japan

In 1982, The Japan Atomic Energy Research Institute completed construction of a dry spent nuclear fuel storage facility at Tokai, Japan for the storage of JRR-3 research reactor spent nuclear fuel (Shirai et al., 1991). The facility consists of a building enclosing several support areas (cask receipt, loading, cask maintenance, and control room) and the drywell storage structure (Figure F-30). The storage structure is 12 m (39.4 ft) long, 13 m (42.7 ft) wide, 5 m (16.4 ft) high concrete box that encapsulates a 10 x 10 lattice array of drywells (Figure F-31). Each drywell storage canister (Figure F-32) comprises a

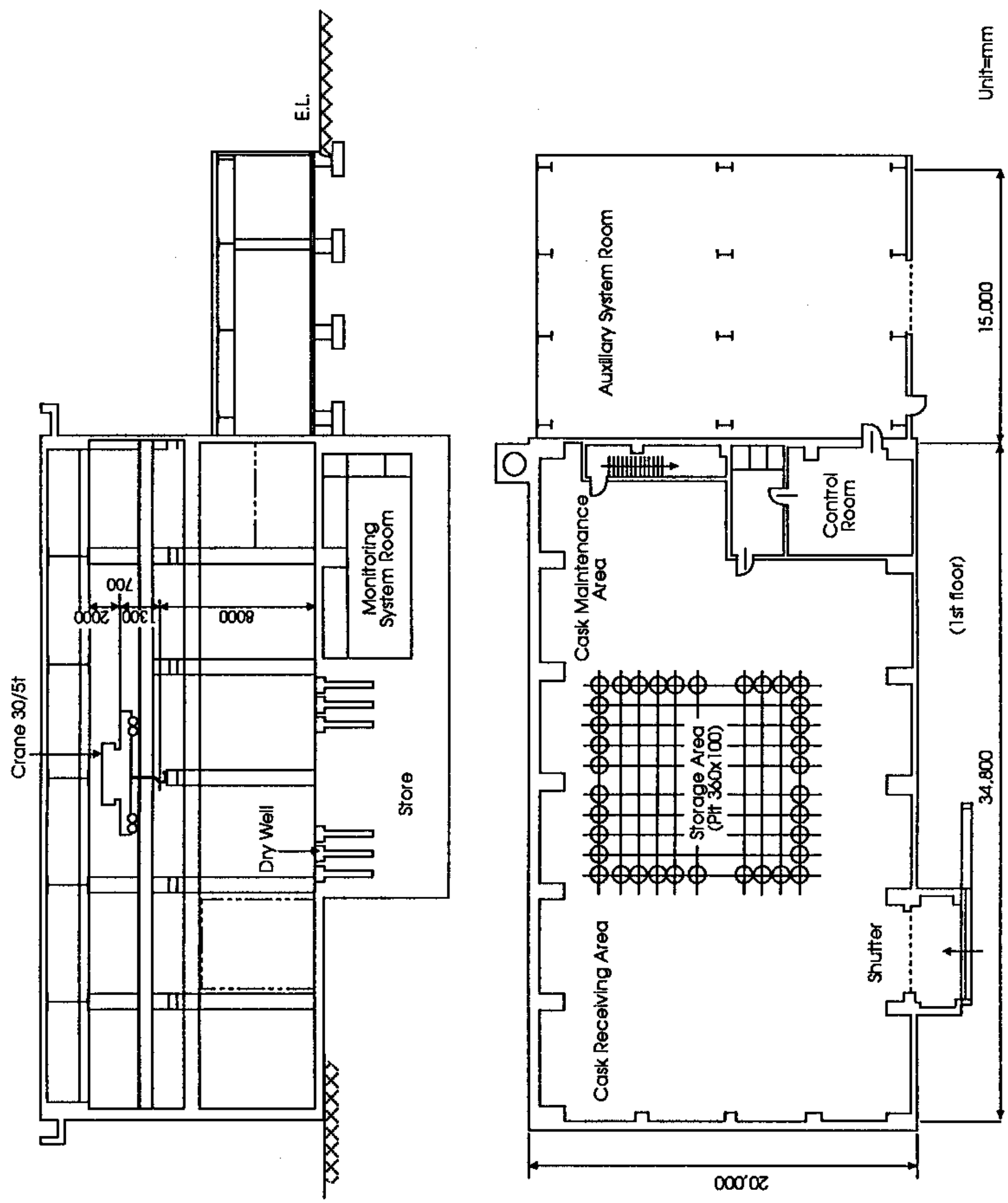


Figure F-30 General Arrangement of Dry Storage Facility at Tokai, Japan

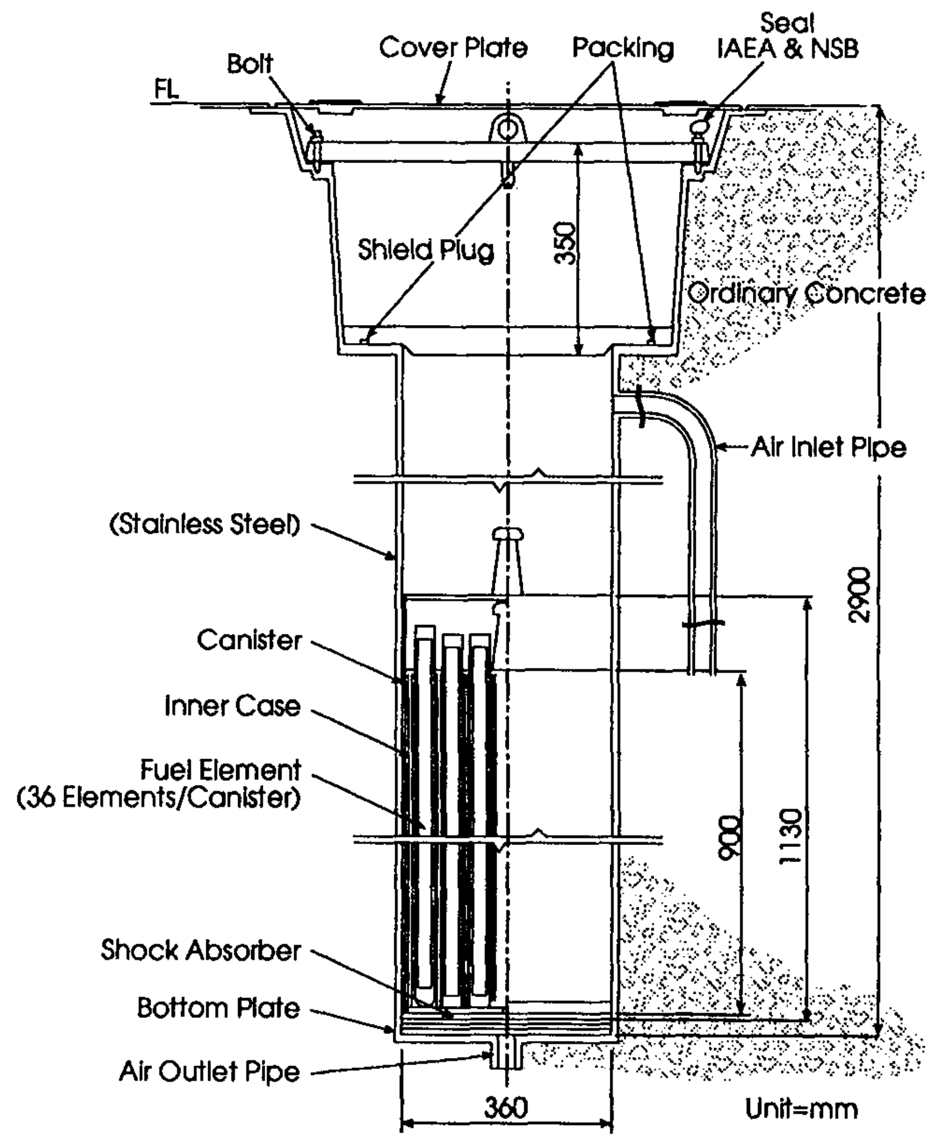


Figure F-31 JRR-3 Dry Storage Facility

0.8 cm (0.3 in) thick stainless steel liner 2.5 m (8.2 ft) deep and has a 36 cm (14.2 in) inner diameter. Each drywell has a labyrinth air inlet and outlet pipe for radiation monitoring and decay heat removal, and is covered with a 35 cm (13.8 in) thick carbon steel shield plug. The plug is bolted to the concrete and has a cover plate above it. Each drywell has a minimum of 1.5 m (4.9 ft) of concrete shielding around it.

A cylindrical stainless steel canister (Figure F-32) is placed in each drywell. The canister has 0.5 cm (0.2 in) thick walls, a 35 cm (13.8 in) outer diameter, and a height of 1.25 m (4.1 ft). Each canister holds 36 fuel elements and is fusion welded after being loaded with spent nuclear fuel, evacuated, and filled with inert gas. Each element is a natural metallic or 1.5 percent ^{235}U-enriched uranium oxide cylinder encased

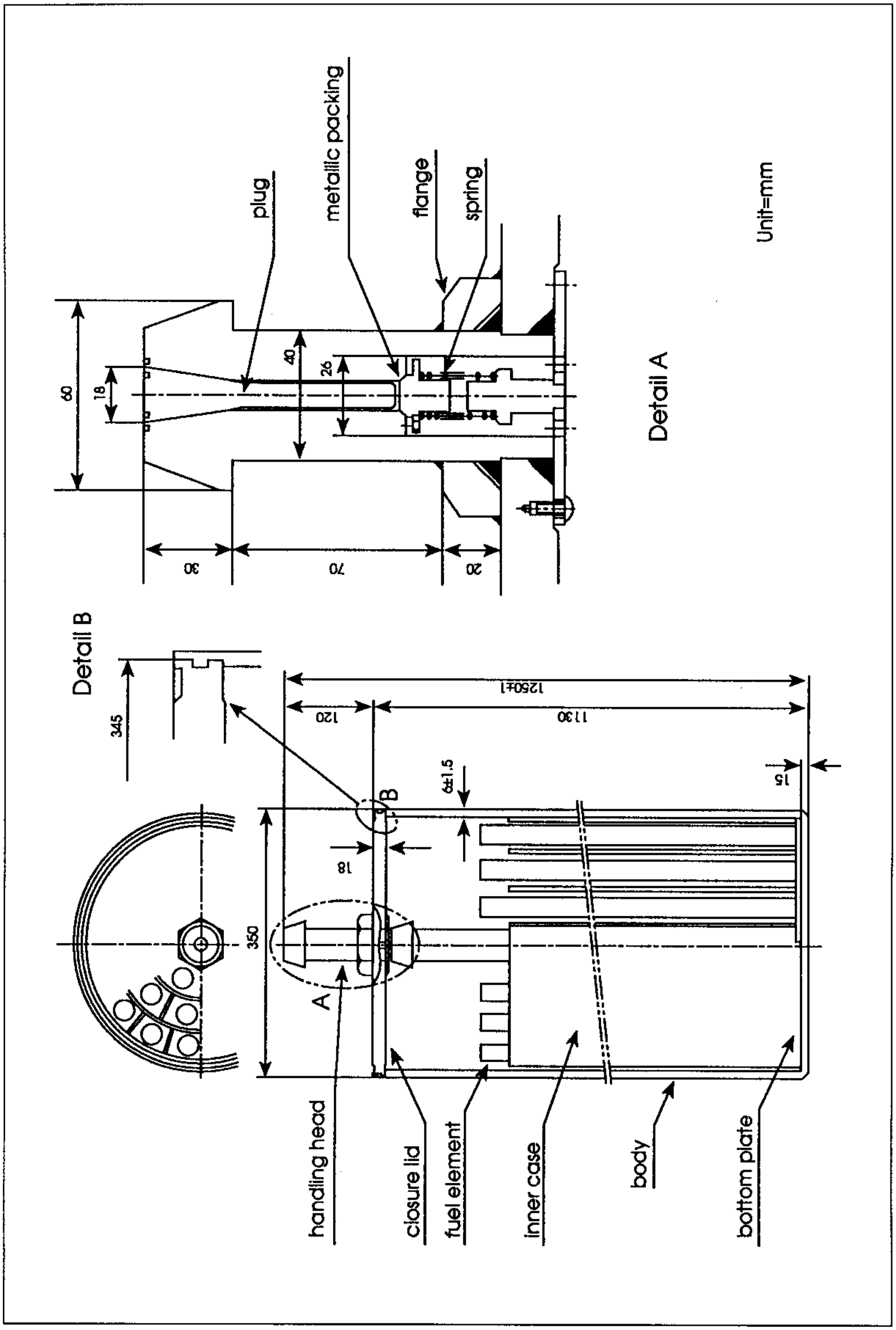


Figure F-32 Storage Canister

in aluminum cladding. The element is about 95 cm (37.4 in) long, with a 2.5 cm (1 in) outer diameter. Before storage, each element must meet the specifications of:

- maximum burnup = 800 Megawatt Days per metric ton,
- minimum cooling time = 2,500 days (6.8 years),
- maximum fission product activity = 110 Ci, and
- maximum decay heat = 0.5 Watt.

Although the low decay heat for the fuel elements eliminates the need for any cooling system (i.e., natural convection and conduction are sufficient), a system of blowers, filters, dehumidifiers, and monitors is provided for the facility. This system is used to provide subatmospheric pressure in the drywell so that any leakage would always be into the structure and its associated filters. This system ensures low humidity for minimizing any corrosion, and is also part of the radiation monitoring system designed to sample airflow for any escaping Krypton-85, a long-lived fission product that would be indicative of canister and fuel degradation. Maximum fuel storage temperature is maintained below 45°C (113°F) in this storage facility.

As of 1991, 1,800 fuel elements [14 metric tons (15 tons) of uranium] have been stored at this facility without any incidents. In 1987, after 5 years of storage, 2 canisters and their 72 fuel elements were removed and examined. None of the fuel elements or canisters exhibited any signs of corrosion, cracking, degradation, or failure.

F.2.10 Summary

In summary, the preceeding discussion indicates the following:

- Dry storage of spent nuclear fuel is a mature technology and requires the least maintenance.
- Dry storage of foreign research reactor spent nuclear fuel appears to be practical using existing designs from commercial utility experience and has been demonstrated in operating storage facilities for foreign research reactor spent nuclear fuel in Australia and Japan.
- Wet storage technology is the most common method for spent nuclear fuel.

F.3 Selection of Storage Technologies for Further Evaluation

The preceding discussions have identified several storage technologies suitable for foreign research reactor spent nuclear fuel. Three basic categories encompass these technologies:

- Dry Vault Storage,
- Dry Cask Storage, and
- Wet Pool Storage.

The three technologies are discussed in this section, including site-specific modifications, while Section F.4 describes the potential impacts and ramifications at the five candidate management sites. All three approaches are estimated to require less than 4.5 ha (11 acres) of site land for receipt of all of the foreign research reactor spent nuclear fuel under consideration in this EIS.

F.3.1 Dry Storage Facility Designs

F.3.1.1 Spent Nuclear Fuel Storage Using a Dry Vault (Modular Dry Vault Storage)

As noted previously, the dry vault facility is an aboveground, self-contained concrete structure that includes dry fuel loading and unloading (Fort St. Vrain, 1992; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993). The vault approach design consists of four major components: a receiving/loading area, fuel storage canisters, a shielded container handling machine, and a modular array for storing the fuel storage canisters (Figure F-33). The receiving area uses a small wet pool for unloading the transportation casks and for short-term (1 to 3 year) storage of foreign research reactor spent nuclear fuel exceeding 40 Watts per element. Table F-17 summarizes some typical modular dry vault storage parameters. The vault consists of several array units, and each unit provides storage for hundreds of fuel elements. The vault itself consists of a charge/discharge bay with a fuel handling machine above a floor containing steel tubes that house the (removable) fuel canisters. Shielding above the spent nuclear fuel is provided by the thick concrete floor and shield plugs inserted into the top of the steel storage tubes. The steel tubes serve as secondary containment for the foreign research reactor spent nuclear fuel and descend into an open storage area. Large labyrinth air supply ducts and discharge chimneys permit natural convection cooling of the steel spent nuclear fuel storage tubes, while the perimeter concrete walls provide for shielding. The design allows for expansion by adding additional units of arrays to the end of the vault, or by construction of another module. The vault facility also includes a receiving and loading bay that allows handling of the shielded transportation casks and unloading of the foreign research reactor spent nuclear fuel into the short-term wet storage pool. The receiving bay provides for spent nuclear fuel inspection, canning as required, and could be used for spent nuclear fuel characterization with additional equipment and modifications.

In operation, the transportation cask is lifted by a crane and placed in the unloading area of the small wet pool. The fuel elements are removed under water, examined, and, if the heat generation rate is below 40 Watts per element, the spent nuclear fuel is placed within the transfer canister. The transfer canister is subsequently drained, dried, and seal-welded. The handling machine then transports the loaded canister to the storage tubes. The handling machine includes radiation shielding. Heat dissipation is accomplished by natural convection from the surfaces of the handling machine and canister. The handling machine transfers the spent nuclear fuel canister from the receiving area to the vault and places the canister vertically into the storage tubes. The shield plug is placed on top of the loaded storage tube. Decay heat is dissipated by natural convection; air enters through inlet ducts at the bottom of the vault module, passes around the outside of the steel storage tubes containing the spent nuclear fuel canisters, and exits through outlet ducts at the top of the module. The vault facility stores spent nuclear fuel in canisters that are approximately 40.6 cm (16 in) in diameter by some 4.6 m (15 ft) long. As currently envisioned, foreign research reactor spent nuclear fuel would be stored in 5 levels of fuel with 4 elements per level, for a total of 20 fuel elements per spent nuclear fuel canister (MTR aluminum-clad type design). The vault design allows for 36 to 44 canisters per array unit, depending upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding with an air inlet temperature of 49°C (120°F)] (Shedrow, 1994a; Taylor et al., 1994).

Thus, the number of vault units/arrays required is as follows:

- spent nuclear fuel decay heat exceeding 80 Watts per element: 35 vault units;

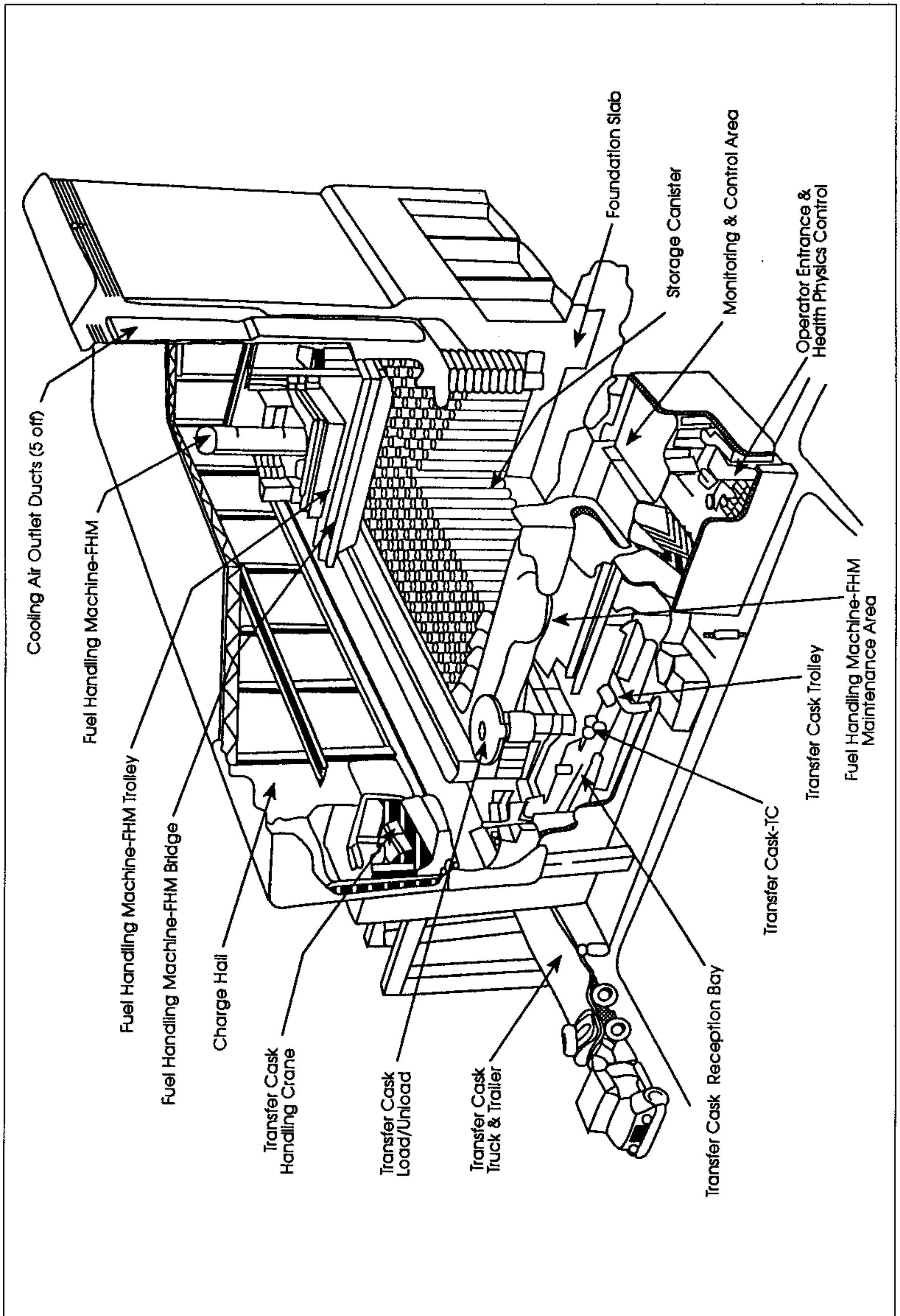


Figure F-33 Vault Elevation View

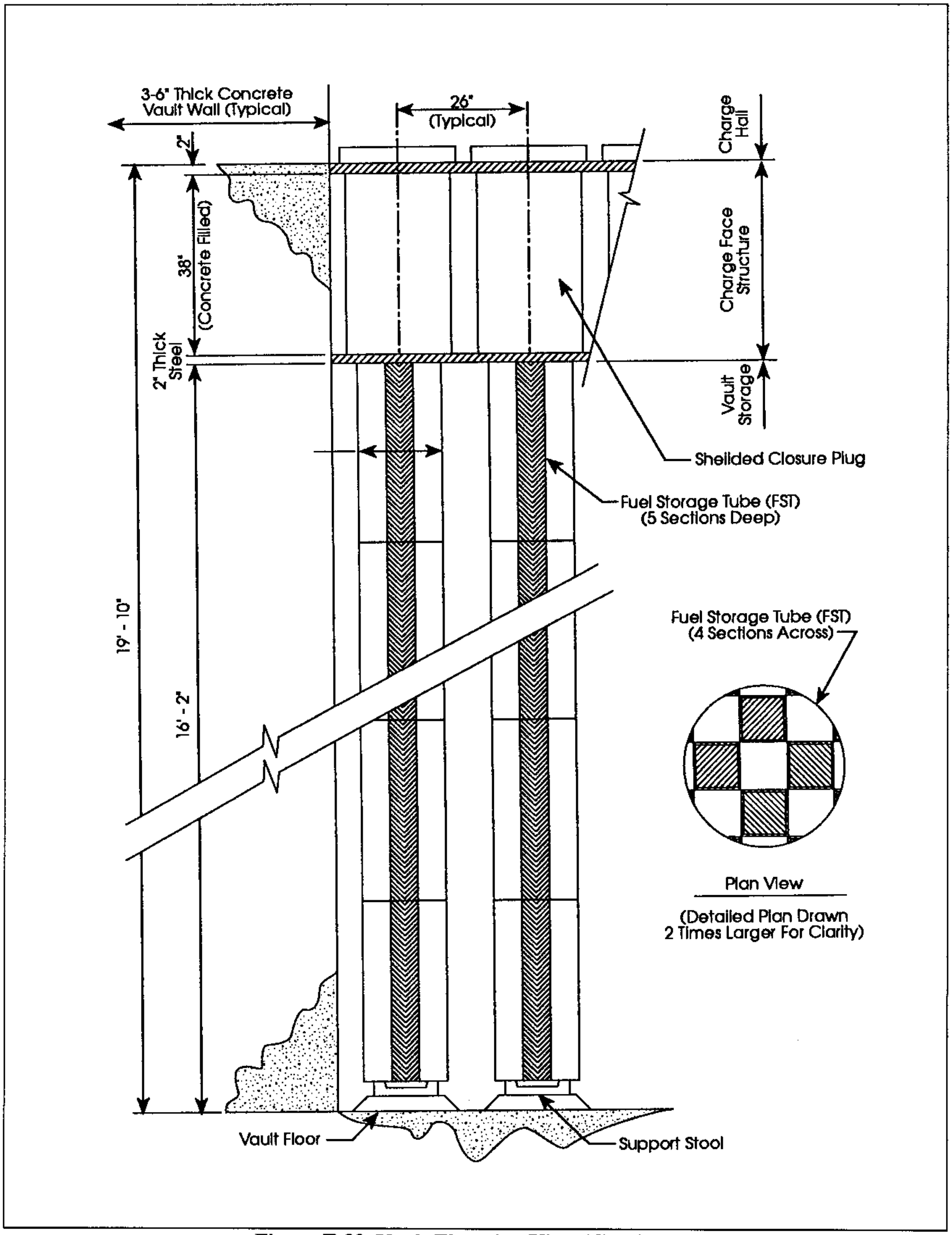


Figure F-33 Vault Elevation View (Continued)

Table F-17 Summary of Modular Dry Vault Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3.7 ha (9 acres)
Facility:	
Size (Area)	5,000 m^2 (54,000 ft^2)
Concrete	21,800 m^3 (28,500 yd^3)
Steel	5,200 mt (5,750 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	835,000 l (221,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	190/yr average, 234/yr peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$370 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/year) for first 10 years, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	22 m^3/yr (780 ft^3/yr) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$15.6 million during handling, $0.6 million during storage[b]

[a] *Staging facility parameters are based upon the regionalized, small wet pool (Dahlke et al., 1994).*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

- spent nuclear fuel decay heat between 40 and 80 Watts per element: 32 vault units; and
- spent nuclear fuel decay heat between 10 and 40 Watts per element: 28 vault units.

For "cold" fuel (10 Watts per element), potentially more than 44 spent nuclear fuel canisters could be placed per vault unit. This would require a customized design. Figure F-34 displays the 10 to 40 Watts and 80 Watts per element cases.

Higher decay heat foreign research reactor spent nuclear fuel would have to be temporarily stored in the small wet storage pool. The storage period is not expected to exceed 3 years.

Criticality concerns are addressed by fuel geometry within the canister and by the use of nuclear poisons (e.g., borated steel in the baskets, etc.). Vault geometry is used to maintain a minimum spacing between adjacent fuel elements or groups of fuel elements to prevent criticality. Nuclear poisons absorb neutrons, thus preventing criticality.

The vault/canyon design has been licensed by the NRC for a specific site. It represents a complete stand-alone facility that can be dedicated to foreign research reactor spent nuclear fuel without requiring the utilization of any other facilities at the host site. Cask handling, maintenance, spent nuclear fuel loading, spent nuclear fuel inspection, spent nuclear fuel storage, and (potentially) characterization can all be accomplished within the same facility.

The cost to construct a modular dry vault storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the vault storage

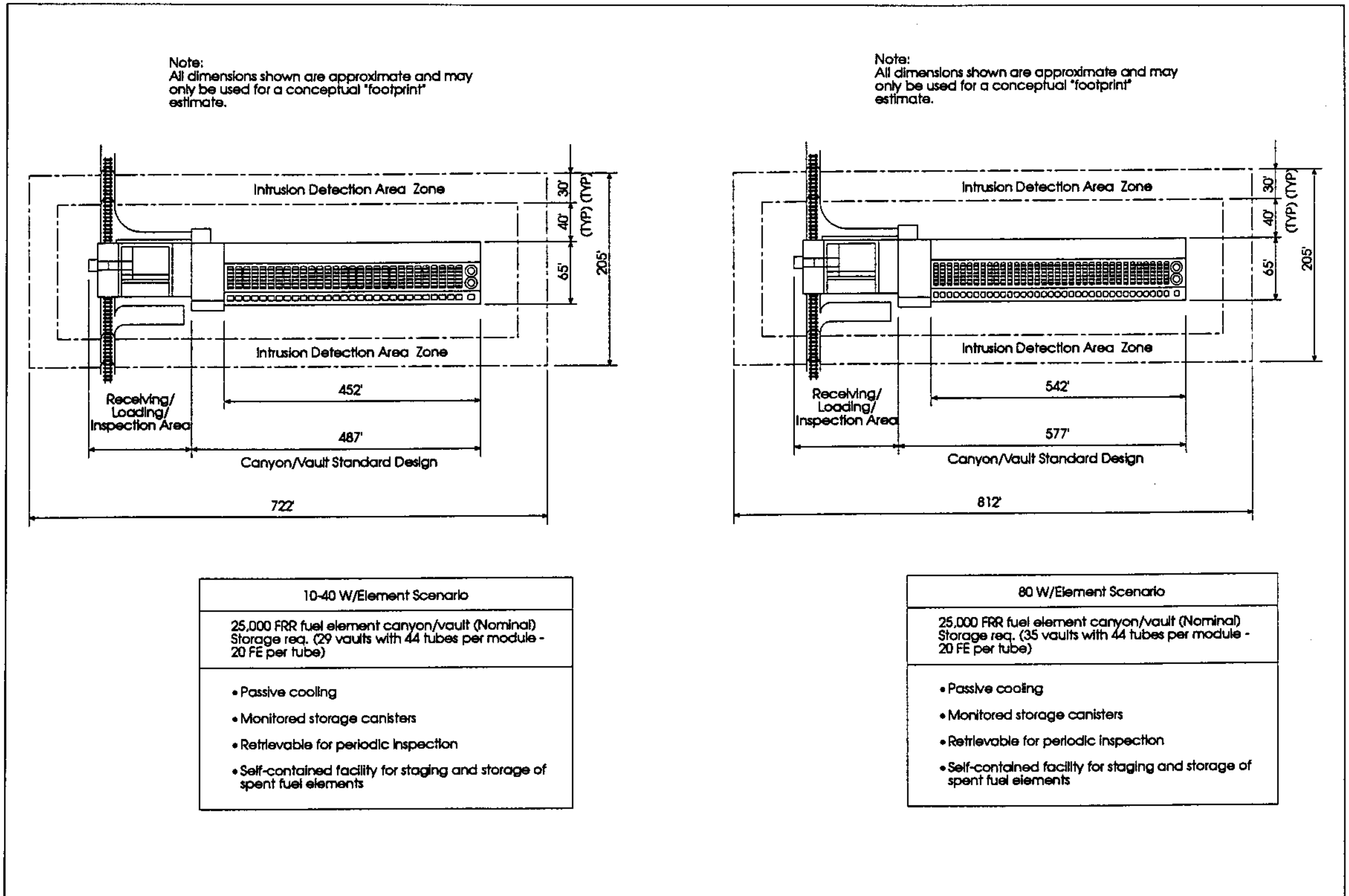


Figure F-34 Canyon/Vault Standard Design (10 to 40 Watt and 80 Watt Scenarios)

area is estimated to be $370 million. The annual operating cost for this facility is estimated to be $15.6 million during the period of handling and transfers of the spent nuclear fuel and $0.6 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.1.2 Spent Nuclear Fuel Storage Using Dry Casks

The dry cask storage approach consists of the following components (BG&E, 1989; Duke Power Company, 1988; Shedrow, 1994a and 1994b; Taylor et al., 1994; Claxton et al., 1993):

- a staging facility for cask receipt and unloading and for loading foreign research reactor spent nuclear fuel into the dry storage casks [a wet pool is used for this purpose, and for short-term (1 to 3 years) storage of foreign research reactor spent nuclear fuel with a heat load exceeding 40 Watts per element],
- an inspection/characterization facility, for examining fuel integrity and canning degraded spent nuclear fuel as required (this may be incorporated into the staging facility as an inspection cell or be immediately adjacent to it),
- a dry storage cask (usually concrete) [this provides for the shielding and the structural stability of the spent nuclear fuel storage],
- a transfer mechanism, such as a dedicated truck/trailer combination with a ram for horizontal modules, or a crane for vertical modules, and
- a separate fuel canister which may or may not be used [if used, it is typically around 4.6 m (15 ft) long and 1.7 m (5.5 ft) in diameter and weighs around 32 metric tons (36 tons)].

The dry cask approach requires the staging facility to receive and inspect the spent nuclear fuel shipment. The transportation cask would be unloaded in a small wet pool within the facility. Subsequently, the spent nuclear fuel is loaded into the dry cask (or spent nuclear fuel canister for the horizontal cask), and the cask is placed upon an outside concrete slab. The horizontal approach uses a dry spent nuclear fuel transfer canister for containing the spent nuclear fuel. This is placed within a shielded transfer cask and moved to the outside modular storage facility. A hydraulic ram inserts the transfer canister inside the horizontal storage module, followed by sealing with a shield plug.

The dry storage modules are designed to withstand normal loads and design basis accident effects, such as earthquakes, tornadoes, and floods. The concrete provides radiation shielding for gamma rays and neutrons. Natural air circulation dissipates the heat as air enters through inlet vents near the bottom of the cask, passes around the spent nuclear fuel canister, and exits near the top. Screens and grills keep birds and other animals out of the cooling duct area. Some of the candidate sites have facilities which may be used for cask receipt and unloading and spent nuclear fuel inspection and transfer to storage.

The application of dry cask storage technology to foreign research reactor spent nuclear fuel depends upon the heat load. Horizontal casks are anticipated to be slightly more restrictive than the vertical casks with respect to the heat load, and are thus the focus of the discussion. The standard design for a horizontal fuel canister provides for 24 or 52 sleeves (i.e., Pressurized Water Reactor or Boiling Water Reactor spent nuclear fuel, respectively), each about 4.6 m (15 ft) long. As with the vault approach, it is conservatively assumed that each sleeve contains five foreign research reactor spent nuclear fuel elements (i.e., in layers), within a basket or can arrangement for maintaining spacings and retrievability. As with the vault approach, the number of dry storage casks depends upon the decay heat of the spent nuclear fuel and a cladding temperature limit [175°C (347°F) for aluminum cladding, with an air inlet temperature of 49°C (120°F)]. The 24-sleeve design allows for a maximum of 120 elements of foreign research reactor spent nuclear fuel with 40 to 80 Watts per element of decay heat, while the 52-sleeve design provides for a maximum of 260 elements per dry storage cask. Thus, the number of casks required is as follows:

- decay heat between 40 and 80 Watts per element: 205 casks, and
- decay heat between 10 and 40 Watts per element: 94 casks.

Note that these values are very conservative and correspond to a maximum of around 40 percent of the NRC-licensed heat loads per cask. Initially, foreign research reactor spent nuclear fuel with higher heat loads could be unsuitable for the dry storage cask pending detailed heat transfer analysis and a determination of limiting fuel storage temperature for aluminum-clad and TRIGA spent nuclear fuel. However, such relatively high decay heat fuel represents a small percentage of the currently identified foreign research reactor spent nuclear fuel so that its impact would be small; and after 1 to 5 years of wet storage, it would all be below a heat duty of 80 Watts per elements. The storage approach assures a minimum spent nuclear fuel wet storage time of 3 years after discharge prior to dry storage. This would essentially ensure that all foreign research reactor spent nuclear fuel is below a heat output of 40 Watts per assembly.

Figure F-35 displays approximate layouts for the dry cask storage facility predicated upon a horizontal cask design. Table F-18 summarizes some general parameters of dry cask storage.

The dry storage cask technology requires a separate staging facility for foreign research reactor spent nuclear fuel unloading, canning, and storage cask loading and transportation cask maintenance. This facility has the following operational areas:

- *Transportation Cask Handling*: This incorporates cask maintenance, truck/railcar unloading, decontamination/washdown, radioactive material control, and cask sampling/flushing/degassing.
- A small wet pool for fuel transfer and short-term storage.
- *Spent Nuclear Fuel Unit Handling*: Fuel removal, decontamination, fuel drying, fuel canning, inerting, and thermal measurements.
- *Spent Nuclear Fuel Unit Transfer*: This constitutes placement of the spent nuclear fuel into the cask or canister, followed by sealing.
- *Radwaste Treatment*: This includes collection, treatment, and preparation for disposal of contaminated effluents and radwaste treatment and solidification.
- *Heating, Ventilation, and Air Conditioning*: This represents heating, ventilation, and air conditioning of the facility so that contamination of the workers and the environment is avoided.

The inspection/characterization facility includes a shielded dry hot cell for spent nuclear fuel analysis and examination, and canning of degraded spent nuclear fuel. All equipment and instrumentation within the cells is remotely operated to provide chemical, physical, and radiological properties, as needed. The facility is maintained under negative pressure with exhaust through High Efficiency Particulate Air filters

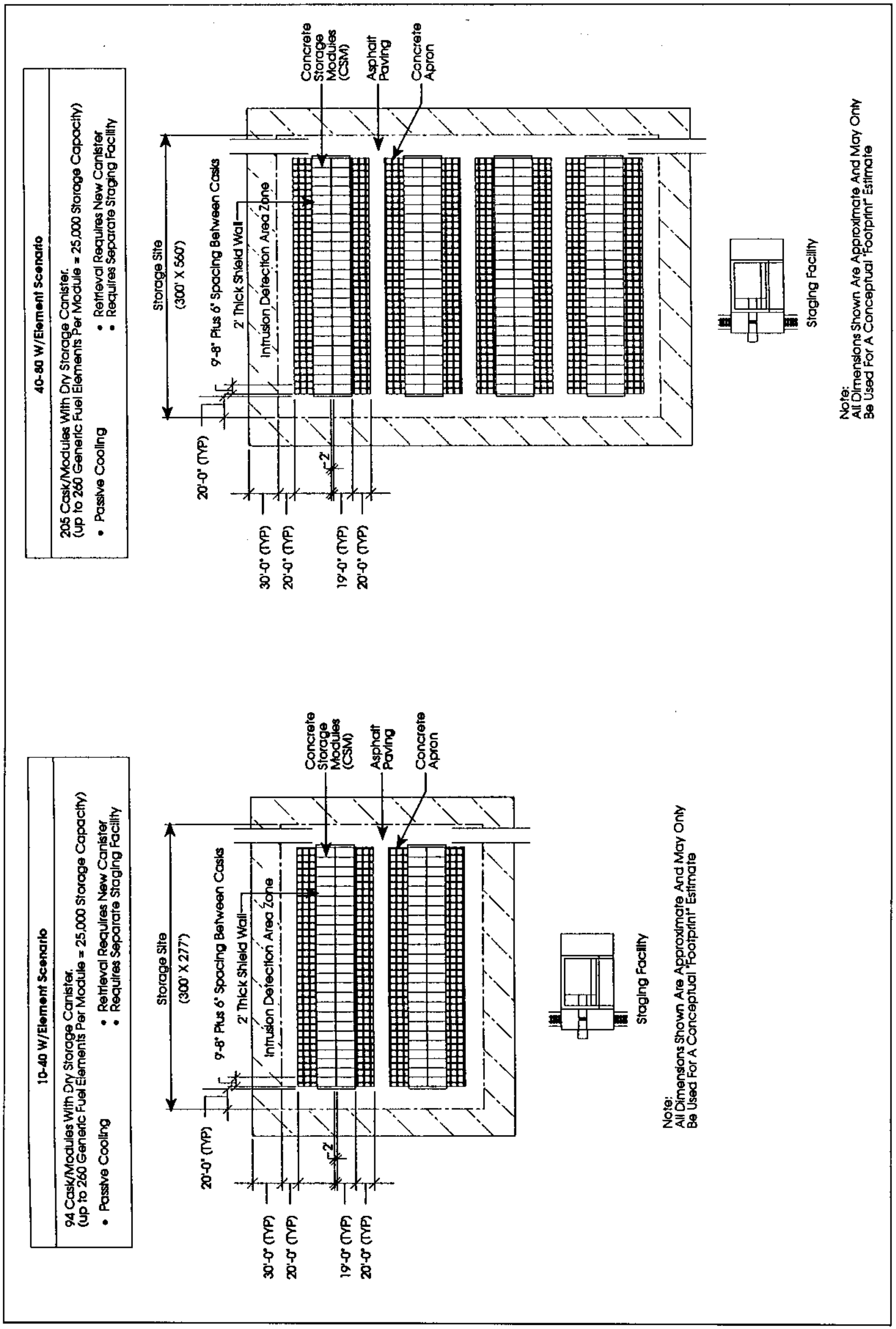


Figure F-35 Schematic of Expanded Dry Cask Storage Facility (10 to 40 Watts Scenario and 40 to 80 Watts Scenario)

Table F-18 Summary of Dry Cask Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel[a]

Construction Phase:	
Disturbed Land Area	3 ha (7.7 acres)
Facility: Size (Area) Concrete Steel	 2,200 m^2 (24,000 ft^2) 17,500 m^3 (22,900 yd^3) 4,500 metric tons (5,000 tons)
Soil Moved	11,000 m^3 (14,400 yd^3)
Equipment Fuel	810,000 l (214,000 gal)
Construction Debris/Waste	1,800 m^3 (2,400 yd^3)
Work Force	50/yr for staging facility, 50 per 25 cask array, 1 array/yr
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$366 million[b]
Operation Phase:	
Electricity	800 - 1,000 MW-hr/yr (staging facility)
Water	2.1 million l/yr (550,000 gal/yr) during receipt, 0.9 million l/yr (238,000 gal/yr) thereafter
Wastestreams	
Solid Low Level Waste	16 m^3/year (565 ft^3/year) during receipt, 1 m^3/yr (35 ft^3/yr) thereafter
Waste Water	1.58 million l/yr (412,000 gal/year) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30 during receipt, 8 thereafter
Annual Operating Cost	$17.3 million during handling, $0.3 million during storage[b]

[a] *Staging facility parameters based upon the Regionalized, Small Wet Pool (Dahlke et al., 1994)*

[b] *Cost estimates are in $1993 (EG&G, 1993)*

to mitigate the environmental effects of any radionuclide releases. This facility is normally located immediately adjacent to, or within, the staging facility.

Dry cask storage is unique among the three storage technologies because of its ability to be operationally integrated with existing facilities, which allows for faster implementation as compared to the other two storage technologies. Several DOE sites have facilities with spent nuclear fuel handling capabilities similar to the requirements of the staging facility. Potential examples include the RBOF at the Savannah River Site and the ICPP-666 storage pool area. For dry cask storage, the spent nuclear fuel would be shipped to the existing facility and unloaded from the transportation cask. The spent nuclear fuel would be inspected, canned if identified as a degraded element, and placed inside the storage canister. Spent nuclear fuel with heat loads exceeding 40 Watts per element would be stored in the existing facility to allow cooldown prior to cask storage. After filling, the canister would be sealed and placed inside the storage cask. The only new construction required would be the concrete storage pad (for vertical casks) or the concrete storage modules (for horizontal casks). For the foreign research reactor spent nuclear fuel receipt rate of approximately 2,000 elements per year considered in the analyses in this EIS, approximately 8 storage casks would be needed annually.

The cost to construct a dry cask storage facility with a staging area sufficient to unload, characterize, can, temporarily store in a small pool, and transfer the spent nuclear fuel to the cask storage area is estimated to be $366 million. The annual operating cost for this facility is estimated to be $17.3 million during the period of handling and transfers of the spent nuclear fuel and $0.3 million during the period of storage.

The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993) with the addition of the cost of a small wet storage facility reported by Dahlke et al. (Dahlke et al., 1994).

F.3.2 Wet Storage Facility

Three generic wet storage facility options have been proposed for foreign research reactor spent nuclear fuel. They are denoted Centralized-Underwater Fuel Storage Facility, Regionalized Large-Underwater Fuel Storage Facility, and Regionalized Small-Underwater Fuel Storage Facility (Dahlke et al., 1994). The difference between these 3 options is that Centralized-Underwater Fuel Storage Facility is sized to store 100 percent of the foreign research reactor spent nuclear fuel under consideration in this EIS (Figure F-36), Regionalized Large-Underwater Fuel Storage Facility is designed for the storage of 75 percent of the foreign research reactor spent nuclear fuel, and Regionalized Small-Underwater Fuel Storage Facility will accommodate 25 percent of the foreign research reactor spent nuclear fuel. These three options were selected to encompass any conceivable decision regarding centralization or regionalization (by geography or fuel type) for the foreign research reactor spent nuclear fuel storage sites. The design features of all three wet storage facility options are identical with the exception that building and pool sizes and, in the case of the Regionalized Small-Underwater Fuel Storage Facility, the number of storage pools and receiving bays is smaller for the Regionalized Large-Underwater Fuel Storage Facility and Regionalized Small-Underwater Fuel Storage Facility. Table F-19 presents the difference in design between these three facilities. Because the design and environmental impacts of the larger Centralized-Underwater Fuel Storage Facility would bound the two smaller facility designs, the balance of the presentation in this section addresses the specific design of the Centralized-Underwater Fuel Storage Facility for storage of 100 percent of the foreign research reactor spent nuclear fuel.

The proposed new wet storage facilities consist of a fuel storage area and support areas (Dahlke et al., 1994). The Fuel Storage Area provides for the receipt of cask transportation vehicles, cask unloading and decontamination, fuel handling, transfer, and storage. Support areas provide for the equipment necessary to maintain and operate the storage area (e.g., heating, ventilation, air conditioning, water treatment, and waste management). The wet storage facility would be constructed as a structure that meets all current nuclear regulations for withstanding natural events such as seismic, tornado, and flood, as well as aircraft impact loads. All systems supporting the operation of the fuel storage facility would also meet these safety requirements. The facility is equipped with a 118-metric ton (130-ton) overhead cask handling crane, and a 9-metric ton (10 ton) fuel handling crane. Each cask transportation vehicle would enter the facility through one of two bays, where it would be monitored and washed from transportation dust. When the external surfaces are cleaned, the cask would be placed into a decontamination room where the cask would be prepared as needed to facilitate underwater unloading. The cask would then be placed in an unloading pool. Transportation casks would be monitored and, if clean of radioactive contamination, placed in an unloading pool. The cask receiving area can accept two simultaneous shipments on 3 m (10 ft) by 24.4 m (80 ft) trucks or railcars and casks weighing up to 114 metric tons (126 tons), each with a total individual cask and transport vehicle weight of 177 metric tons (195 tons).

There are two stainless steel-lined unloading pools, one measuring 6.4 m (21 ft) long, 5.8 m (19 ft) wide by 13.4 m (44 ft) deep, and the other measuring 6.1 m (20 ft) long, 6.1 m (20 ft) wide, and 11 m (36 ft) deep. There are two decontamination hot cells. Each unloading pool has a cask washdown system. Prior to being placed in one of the two storage pools, each fuel element would be checked to ensure that it is properly configured for direct transfer to the fuel storage pool buckets. If not, it would be transferred to the fuel cutting/canning pool, which is 10.4 m (34 ft) long, 5.8 m (19 ft) wide, and 9.4 m (31 ft) deep. Here it would be prepared for transfer to the storage pool buckets.

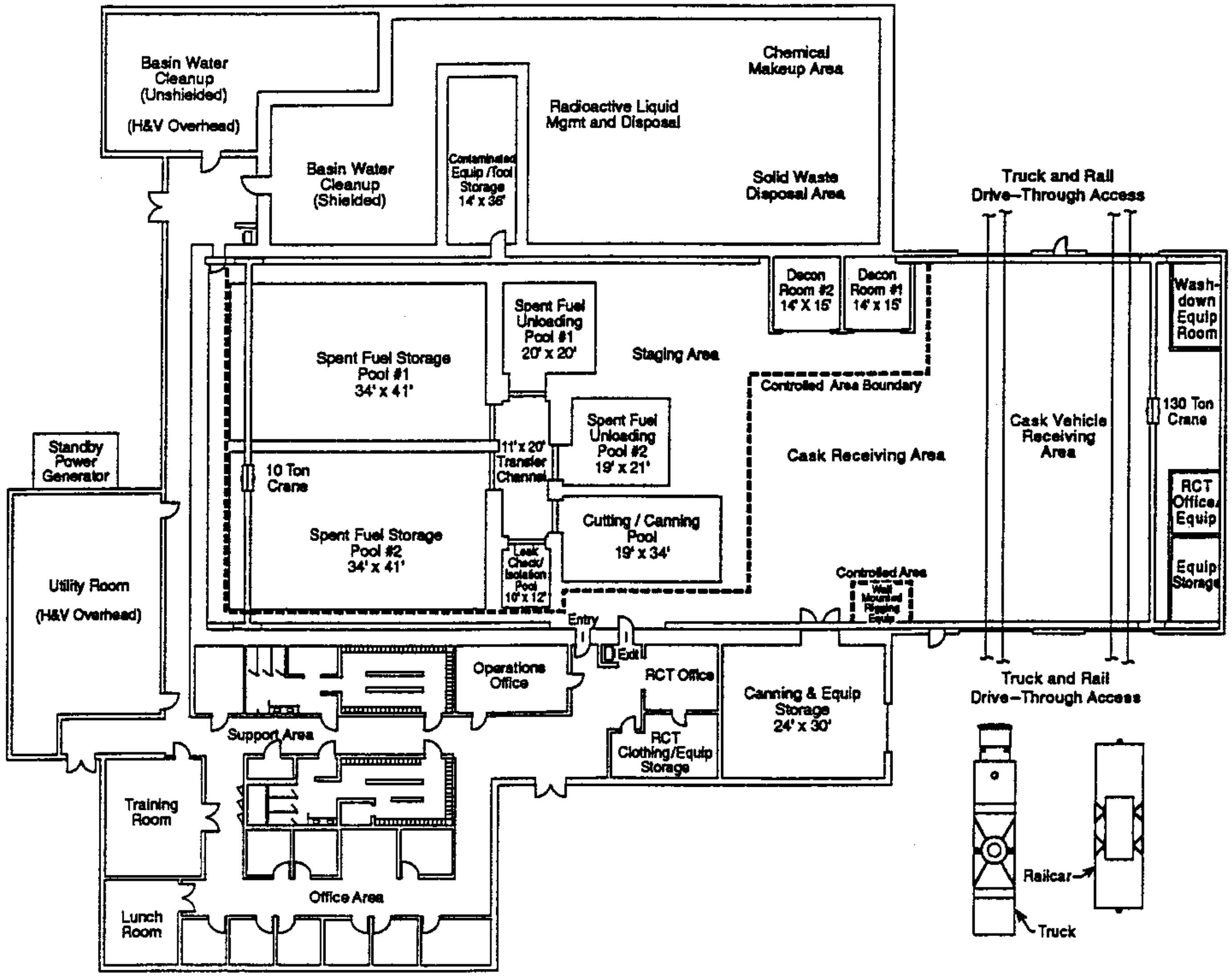


Figure F-36 Generic Wet Storage Facility for All of the Foreign Research Reactor Spent Nuclear Fuel

Table F-19 Design Difference Between 100 Percent, 75 Percent, and 25 Percent Generic Wet Storage Facilities

Design Parameter	Wet Storage Capacity (Amount of Foreign Research Reactor Spent Nuclear Fuel)		
	100%	75%	25%
Number of Storage Pools	2	2	1
Storage Pool Length and Width, m (ft)	16.5 x 10.4 (54 x 34)	12.5 x 10.4 (41 x 34)	10.4 x 8.2 (34 x 27)
Transfer Channel Length and Width, m (ft)	6.1 x 3.4 (20 x 11)	6.1 x 3.4 (20 x 11)	6.1 x 3 (20 x 10)
Fuel Unloading Pool Length and Width, m (ft)	6.4 x 5.8 (21 x 19)	6.4 x 5.8 (21 x 19)	6.1 x 6.1 (20 x 20)
Number of Receiving Bays	2	2	1

If cask measurements indicate that fuel is degraded, the fuel would be transferred to the isolation pool which is 3.7 m (12 ft) long, 3 m (10 ft) wide, and 9.4 m (31 ft) deep. This pool is equipped so that wet sipping, dry sipping, or vacuum sipping of the suspect fuel element could be performed. Sipping is a method of measuring radioisotope leakage from spent nuclear fuel. An identified degraded fuel element would then be transferred to the cutting/canning pool, where it would be canned before transfer to the storage pool. If it is not found to be degraded, it would be transferred directly to the storage pool.

All six pools in this facility (two unloading; two storage, cutting/canning; and two leak check/isolation) are hydraulically connected by a stainless steel-lined transfer channel/pool which is 6.1 m (20 ft) long, 3.3 m (11 ft) wide, and 9.4 m (31 ft) deep. Gates between this transfer channel and each pool allow for hydraulic watertight isolation of the other pools to control contamination and allow for individual pool water pump-out. All pools and channels are constructed of concrete with stainless steel floors and liners. Pool water leak detection and collection systems are provided in accordance with NRC Regulatory Guide 1.13 (NRC, 1975) and American National Standards Institute Standard N305-1975 (ANSI, 1975b).

Each of the two stainless steel-lined, interconnected storage pools is 16.5 m (54 ft) long, 10.4 m (34 ft) wide, and 9.4 m (31 ft) deep. Each contains stainless steel storage racks which hold 1,000 fuel storage holes, with a 0.2 m (8 in) spacing maintained between adjacent storage holes (Figure F-37). Each storage hole can hold three stacked stainless steel fuel storage buckets (Figure F-38), which can each contain up to four fuel elements. A loading fixture is used during spent nuclear fuel emplacement. Thus, each pool has the capacity for 12,000 fuel elements (or 24,000 for both pools). The 0.2-m (8-in) space provides neutron isolation between adjacent storage holes and, therefore, ensures criticality safety. Each rack is 2 m (6.7 ft) square and 3.2 m (10.4 ft) high, and consists of a 5 x 5 array of 25 fuel positions. A hinged lid is above each of these fuel positions. Each pool can hold 40 of these racks. Fuel elements are stored in the racks so that at least 25.4 cm (10 in) of rack protrudes above the top of the fuel.

The heating, ventilation, and air conditioning system for the wet storage facility would include a room for air supply equipment and a room for air exhaust equipment with separate filtering and monitoring rooms for the different areas within the facility. There are two heating, ventilation and air conditioning equipment rooms. Although a total of six different rooms are used for heating, ventilation, and air conditioning, all exhaust air is directed through pre-filters, High Efficiency Particulate Air filters, radiation monitors, filter fire protection components, and heat recovery coils before it exhausts to the atmosphere.

The wet storage facility's water treatment system consists of redundant pumps, piping, filters, deionizers, and ultraviolet microorganism control systems. A heat removal system is sized to maintain the bulk water temperature at or below 43°C (109°F). The system's filters and deionizers include anion and cation exchangers that maintain water chemistry and remove radionuclides from the pool water.

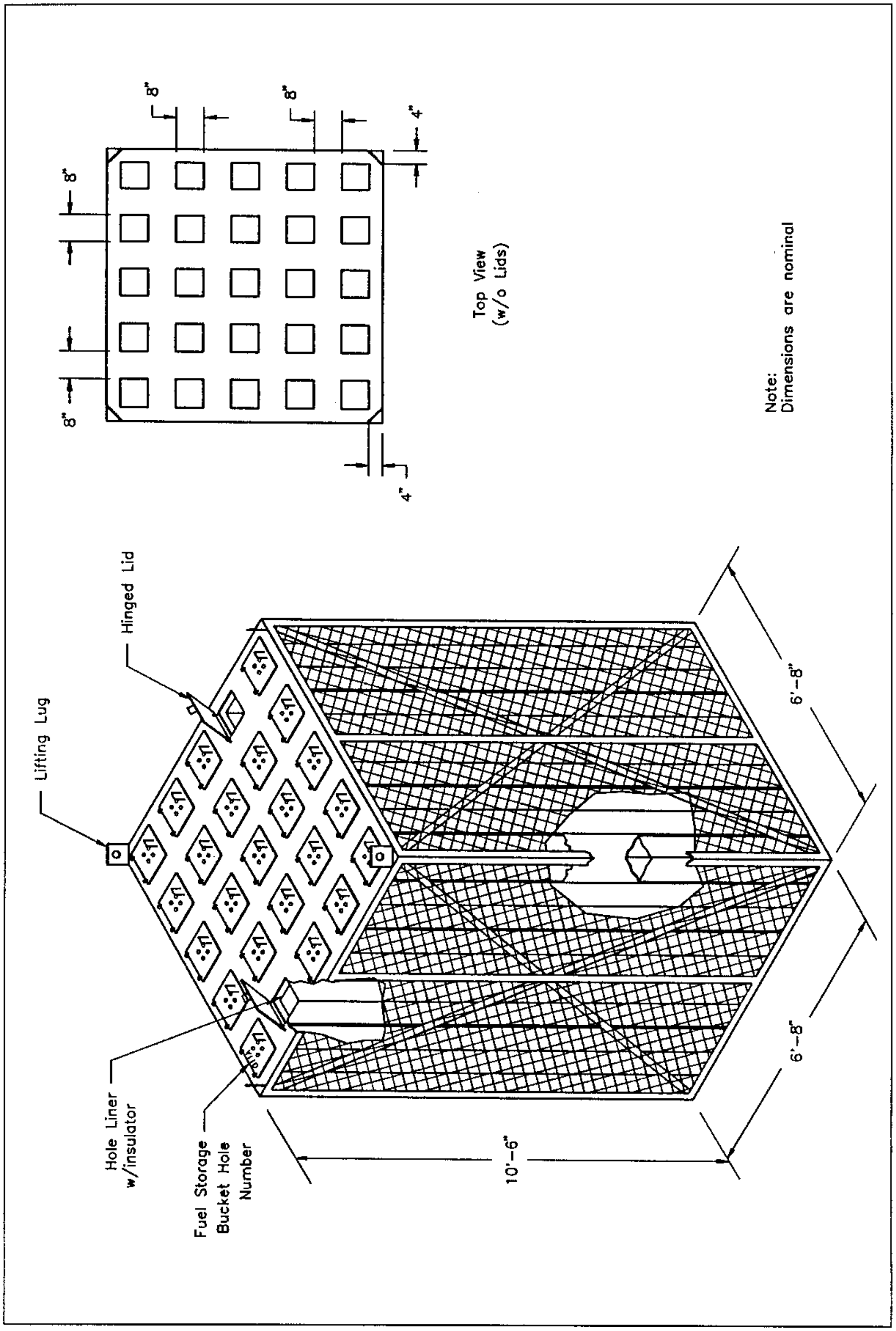


Figure F-37 Storage Racks

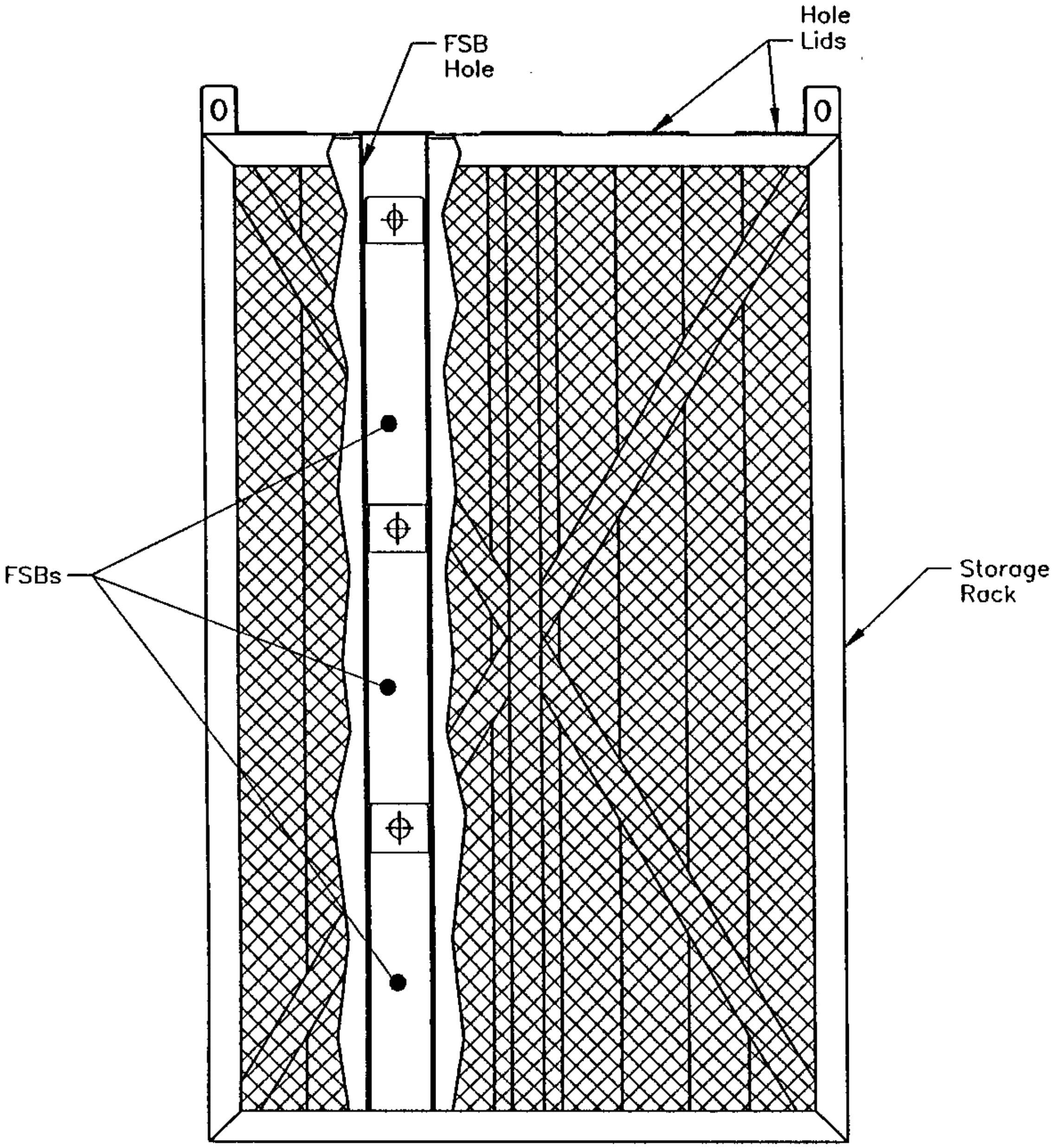


Figure F-38 Stacked Fuel Storage Buckets in Storage Rack

The staff required to operate the wet storage facility is estimated to be a maximum of 30 when 24-hour-a-day fuel loading is being performed, with only occasional maintenance visits by administrative personnel for operation. Potential radiological consequences are extrapolated from other operating wet storage facilities and are discussed in Section F.4.

No high-level radioactive waste is expected to be generated by the wet storage facility. Low-level solid radioactive waste generated over the 40-year life of the facility is expected to be about 488 m^3 (17,200 ft^3). Nonradioactive solid waste generated over the facility's life is expected to be about 300 m^3 (10,594 ft^3). No nonradioactive air emissions are expected to be generated by this facility. Table F-20 summarizes the parameters for the facility.

The cost to construct a wet storage facility with a staging area sufficient to unload, characterize, can, and transfer the spent nuclear fuel to the storage area is estimated to be $449 million. This cost may include some duplicate facilities and equipment present in both the staging facility and the rest of the wet storage facility which were costed separately. The annual operating cost for this facility is estimated to be $23.3 million during the period of handling the spent nuclear fuel and $3.5 million during the period of storage. The cost estimate for the facility is based on a cost report prepared by Idaho Inc. (EG&G, 1993).

F.3.3 Site-Specific Facilities Proposed for Foreign Research Reactor Spent Nuclear Fuel Storage and Management

F.3.3.1 Savannah River Site

RBOF

The Savannah River Site has proposed the use of its RBOF, which is also designated as Building 244-H (DuPont, 1983a and 1983b; Shedrow, 1994a and 1994b; WSRC, 1994a; Claxton et al., 1993; DOE 1993c). The RBOF is a 30-year-old steel and concrete block structure that contains several water pools that have been used for the storage of spent nuclear fuel including foreign research reactor spent nuclear fuel since approximately 1964.

The RBOF facility is located in H-Area on 0.8 ha (2 acres) of land about 397 m (1,300 ft) west of the 221-H Canyon building. A railroad track terminates within the facility, and a roadway surrounds it for access by trucks. The RBOF Building (244-H) is about 42 m (139 ft) wide and 45 m (148 ft) long and contains water-filled basins. The basin area extends below grade to a maximum depth of 13.7 m (45 ft), the roof over the 91 metric ton (100 ton) crane bay is about 13.7 m (45 ft) above grade, and most of the remainder of the roof is at an elevation of 4.6 m (15 ft).

The building consists of seven main sections separated by partition and shielding walls. A ventilation system is provided to exhaust any airborne particulate contamination through filters. The basins, cubicles, and shielding walls are made of reinforced concrete. Most of the above-grade structure consists of standard structural steel shapes with an exterior wall of Transite™ (registered trademark of Johns-Manville Co.). The walls are insulated with Fiberglas™ (registered trademark of Owens-Corning Corp.).

The basin, or working area, of the building has an inner wall of Transite™ to prevent water damage to the insulation from condensation. The disassembly and inspection basins are separated by an inner concrete block wall, and the repackaging basins are enclosed by concrete block walls.

Table F-20 Summary of Wet Storage Parameters for Foreign Research Reactor Spent Nuclear Fuel

Construction Phase:	
Disturbed Land Area	2.8 ha (7 acres)
Facility: Size (Area) Concrete Steel	 3,800 m^2 (41,000 ft^2) 12,400 m^3 (16,260 yd^3) 3,100 metric tons (3,443 tons)
Soil Moved	18,000 m^3 (24,000 yd^3)
Equipment Fuel	600,000 l (159,000 gal)
Construction Debris/Waste	2,600 m^3 (10,300 yd^3)
Work Force	157/yr average, 184 peak
Duration (Years)	4 years for construction, 1.5 years for design
Capital Cost	$449 million[a,b]
Operation Phase:	
Electricity	1,000 - 1,500 MW-hr/yr
Water	2.7 million l/yr (720,000 gal/yr) during receipt, 1.5 million l/yr (409,000 gal/yr) thereafter
Wastestreams	
High Level Waste	none
Transuranic Waste (TRU)	none
Solid Low Level Waste	16 m^3/yr (580 ft^3/yr)
Waste Water	1.59 million l/yr (420,000 gal/yr) during receipt, 0.4 million l/yr (109,000 gal/yr) thereafter
Staff (Full-Time Equivalents)	30
Annual Operating Cost	$23.3 million during handling $3.5 million during storage[a]

Source: (Dahlke et al., 1994)

[a] *Cost estimates are in $1993 (EG&G, 1993)*

[b] *The cost may include duplicate equipment costed in both the staging facility and the wet storage facility*

The two storage pools are between 6.7 and 8.8 m (22 and 29 ft) deep, and approximately 70 percent of their capacity is filled with a variety of fuel types, including aluminum-clad fuel with a ^{235}U enrichment up to 93.91 percent. Subcriticality is maintained by appropriate fuel spacing [center-to-center fuel spacing in these racks currently varies between 23 and 65 cm (9 and 25.5 in) depending on the specific rack] in the storage racks, since no neutron absorption material is used in the pool water. Rack height is 3.4 m (11.17 ft), but some fuel protrudes above the top of the racks. Some of this fuel has been stored at the RBOF for as long as 15 years without any significant degradation. These aluminum storage racks have been present in the pools for 30 years without degradation. Figure F-39 shows the floor plan, and Figure F-40 displays an elevation view.

The RBOF includes specific design, operating, and maintenance procedures for the receipt of a wide variety of fuel types and casks, including damaged fuel elements. The RBOF has the facilities and experience in all aspects of spent nuclear fuel receipt including cask wash, fuel unloading, fuel transfer, fuel storage, fuel inspection, fuel disassembly, and fuel repackaging.

The RBOF pools have a stainless steel bottom and epoxy-coated walls. Pool walls are made of reinforced concrete that varies in thickness from 0.9 m (3 ft) at the top of the pool to 2 m (6.5 ft) at the lower elevations, and the pool floor is a stainless steel liner over a 91-cm (3-ft) thick reinforced concrete slab. Most of the pools have a 6.4-mm (0.25-in) thick stainless steel liner on the floor. The disassembly, inspection, and repackaging basins also have a 3.2-mm (0.125-in) thick stainless steel liner on the walls. Pools or basins are connected by transfer canals with underwater doors.

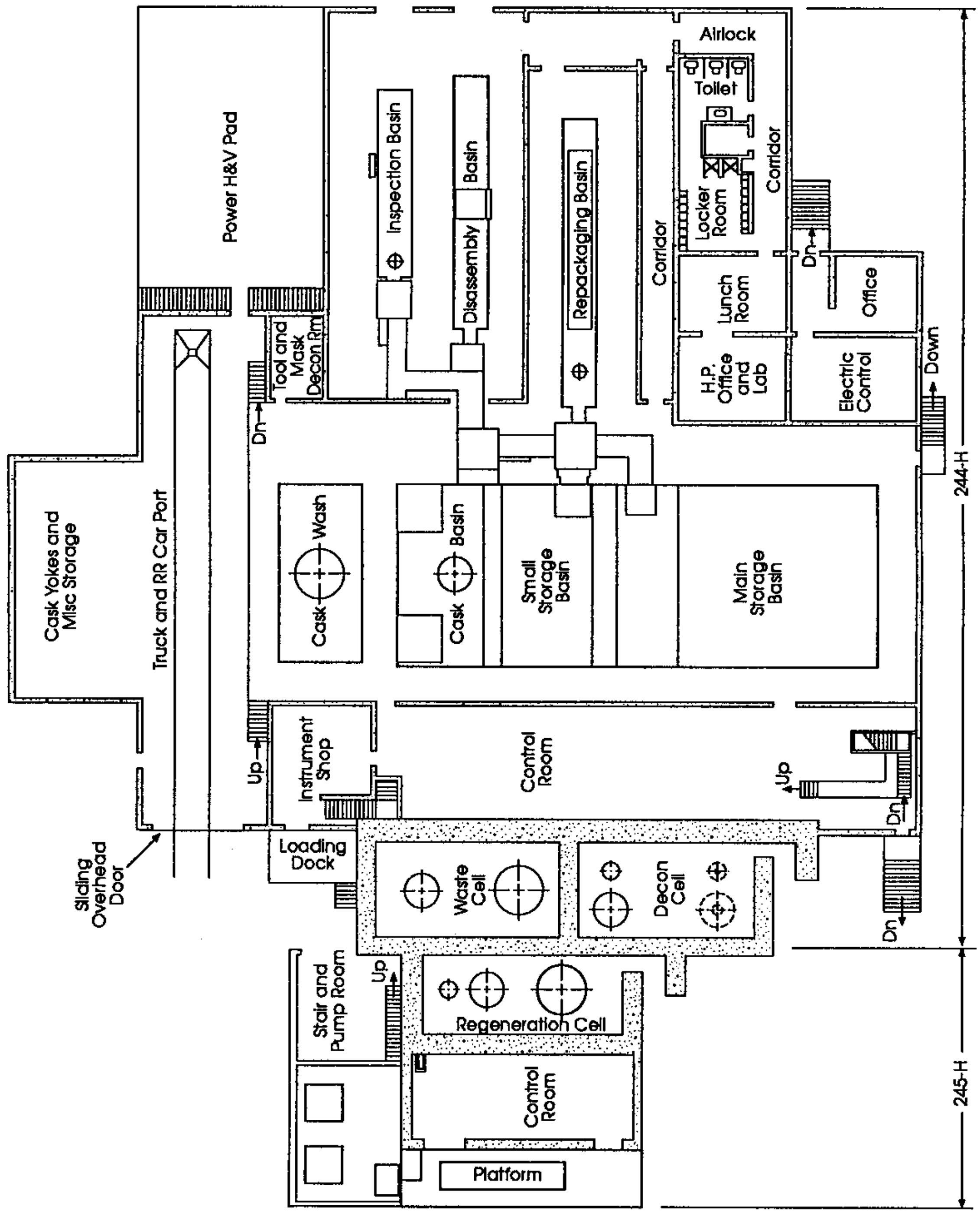


Figure F-39 Plan View of the RBOF

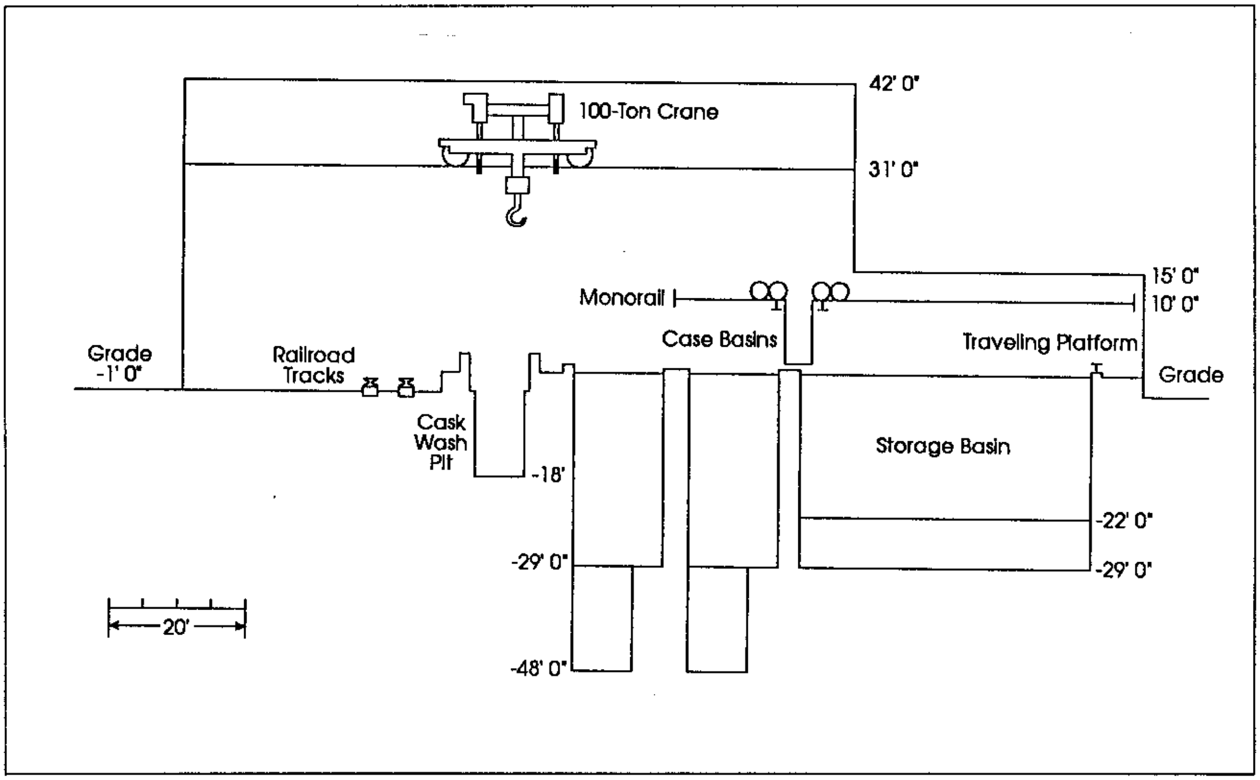


Figure F-40 Elevation Schematic of the RBOF (Facing East)

Water from any basin can be pumped through a filter-deionizer and then returned to the basin as purified water with a conductivity in the range of 0.5 to 1.5 µmhos/cm. In addition, the activity level of the water, which is typically in the range of 0.5 to 1.5 x 10^{-4} µCi/ml, is reduced to less than 0.05 x 10^{-4} µCi/ml by this process. The normal inventory of activity in the approximately 1,700,000 l (450,000 gal) of total basin water is thus 0.1 to 0.3 Ci. For typical flow rates of 454 l/min (120 gal/min), the deionizer processes approximately 1.1 x 10^{8} l (3 x 10^{7} gal) during 6 months of service and may contain about 20 Ci of radioactivity (mostly as cesium) when regeneration is required. The deionizer is typically regenerated every 4 to 5 months. The "Porostone" (aluminum oxide) filter that precedes the deionizer is normally backflushed and recoated with filter-aid whenever a significant pressure drop occurs, which, in practice, is about three to four times per month. When the filter tubes become plugged, they are chemically treated with oxalic acid and sodium hydroxide to open the pores of the filter. This occurs once or twice per year. The purification system maintains excellent chemistry, with mercury and copper kept below two parts per billion, and iron and aluminum maintained below 2 parts per million (ppm). Chloride is maintained below 10 parts per billion.

In the event of an interruption of normal power to the RBOF, critical equipment essential for maintaining personnel safety and containing radioactivity are automatically supplied with emergency power from a 12.5-kilovolt amps, 10-kilowatts, 460-volts gasoline-driven generator.

The building ventilation system serves to minimize airborne radioactivity both inside and outside the facility because of the "once-through" airflow system and the use of High Efficiency Particulate Air filters on the building exhaust. Because of the once-through airflow, activity levels do not tend to increase with time within the building, and the High Efficiency Particulate Air filters serve to effectively remove particulate radioactivity that would otherwise be released to the atmosphere. In addition, the facility is maintained at less than atmospheric pressure so that all building air will be filtered before release. The basin areas are also kept at a lower pressure than the rest of the building, and a separate ventilation system supplies air to the control room, offices, and change room. All air, after filtration, is discharged through a 1.5-m (5-ft) diameter by 16.2-m (53-ft) high stack. The exhaust system for the process vessels in the Waste Cell and the Decontamination Cell is similar to the building system, but is separate and employs acid resistant components. This exhaust is discharged through a 25-cm (10-in) diameter by 16.2-m (53-ft) high pipe.

The 91-metric ton (100-ton) capacity bridge crane travels on a 27.4-m (90-ft) long runway located 9.4-m (31-ft) above grade which permits access to the carport, the cask wash pit, and the cask basin. It is used to handle transportation casks, cask lids, cask basin shims, and a semi-remote impact wrench.

The twin hook crane consists of two 45-metric ton (50-ton) capacity hoist trolleys, which can be arranged for independent travel, or which can be electrically locked to provide for operation as a single unit. Load clearance above the 1.1-m (3-ft 6-in) high cask basin railing is 8.1-m (26-ft 6-in). The bridge crane is pendant-operated from a walkway on the west side of the basins.

A 2.7-metric ton (3-ton) hoist, suspended from a monorail on the south girder of the bridge, is used in the handling of a semi-remotely operated impact wrench. The other 2.7-metric ton (3-ton) hoist is used primarily for the handling of yokes and other ancillary equipment in the yoke storage area adjoining the carport.

Brakes on the cranes and hoist are applied automatically in the event of a power outage.

Two small bridge cranes, one motorized and one manually operated, are employed over the repackaging basin. Both have a load capacity of 2.7 metric tons (3 tons).

The RBOF includes a High Efficiency Particulate Air heating, ventilation, and air conditioning system and maintains subatmospheric pressure within the building to minimize environmental releases of radionuclides. Automatic atmospheric isolation is actuated by activity level monitors inside the RBOF. The RBOF is also equipped with groundwater monitoring for detecting leakage from the pool confinement boundary.

Analysis of the RBOF was performed and included an evaluation of the reliability of process equipment and controls, administrative controls, and engineered safety features. The evaluation identified potential scenarios and radiological consequences. Risks were calculated in terms of 50-year population dose commitment per year (person-rem per year) to the onsite staff and to an individual at the plant boundary. Risk is defined as the product of the expected frequency of a release and the consequences of the release. Consequences are expressed in terms of dose commitment to onsite and offsite populations surrounding the release point.

An evaluation of the RBOF as a potential storage site for foreign research reactor spent nuclear fuel indicates a number of problem areas. The current cask handling capacity of the RBOF is approximately one cask per week. This capacity is based upon facility operations at two shifts per day, 5 days per week. The cask handling capacity could be increased, perhaps to as much as 84 casks per year, if facility operations were expanded to around-the-clock (3 shifts per day), 7 days per week. However, considering that shipments out of the RBOF also require cask handling, the net receipt capacity of the RBOF is practically limited to four casks per month. This capacity would not be sufficient for the potential foreign research reactor spent nuclear fuel cask receipt rate of ~60 casks per year. If the RBOF were used for the receipt and loading of dry storage canisters, its receipt rate could be reduced by half. Only ~1,000 fuel storage spaces are available at the RBOF. Consolidation of the spent nuclear fuel might open an additional 1,425 spaces, but this is much less than that required for the number of foreign research reactor spent nuclear fuel elements under consideration in this EIS. The Savannah River Site has proposed movement of other spent nuclear fuel to the reactor storage basins, and use of dry storage for foreign research reactor spent nuclear fuel.

The DOE Spent Fuel Working Group Report has identified a number of vulnerabilities at the RBOF, including insufficient training, inadequate tornado missile protection, no seismic qualification, lack of water leak detection system, and no up-to-date and approved Safety Analysis Report (DOE, 1993b). It should be noted that a system description and a Safety Analysis Report for the RBOF do exist and were published in 1983. Current recommendations are to address and correct these problems by FY 1996 (DOE, 1993b; Taylor et al., 1994). The 30-year age of these pools may also require analyses to determine the remaining safe lifetime without significant replacement or design modifications.

Reactor Disassembly Basins

Savannah River Site has also proposed the use of one or more of its reactor disassembly basins for Phase 1 storage of foreign research reactor spent nuclear fuel (Shedrow, 1994a and 1994b; Taylor et al., 1994). All of these basins were constructed in the early 1950's and became operational in the mid-1950's. The disassembly basins are similar to each other and are briefly described in the sections that follow, using the L-Reactor disassembly basin as an example.

The L-Reactor performed the basic function of irradiating elements in a heavy water moderated and cooled reactor for the purpose of supplying special nuclear materials for national defense, medical, and research applications. The Savannah River Site production reactors are not currently operating. The disassembly area of the Savannah River Site Production Reactors was designed to serve as a processing area for reactor target and fuel assemblies. This processing included removal of decay heat, disassembly of components, short term storage of fissile product material, and cask loading operations. Total residence time from reactor discharge to shipment to the separation areas was typically 12 to 18 months.

The disassembly basin is arranged into three major sections: the machine basin, the vertical tube storage basin, and the transfer area (Figure F-41). The machine basin and vertical tube storage basins are divided into the following interconnected basins:

Vertical Tube Storage Basin	*Machine Basin*
Deposit and Exit Canal*	Machine Area
Vertical Tube Storage	Horizontal tube storage Target bucket storage Dry cave

* *The Deposit and Exit canal is a water-filled canal that connects the disassembly area and the area that houses the reactor tank top or process room. This canal also acts as a water seal to allow access by an underwater conveyer, but no airflow.*

The disassembly basin contains 12,776,000 l (3,375,000 gal) of water. Vertical tube storage holds 3,730,000 l (985,000 gal), while the machine basin holds 9,050,000 l (2,390,000 gal) (transfer area included). The depth of the disassembly basin ranges from 5.2 to 9.2 m (17 to 30 ft), but most of the basin is around 5.2 m (17 ft) deep. The approximate overall dimensions are 47 m (154 ft) wide by 66 m (216 ft) long. There is a small 0.9 m (3 ft) diameter circular section of the machine basin that is 15.6 m (50 ft) deep.

Figure F-42 shows a basic block diagram of the disassembly process. The reactor assemblies were discharged from the reactor tank to the Deposit and Exit canal, placed in the Deposit and Exit conveyer,

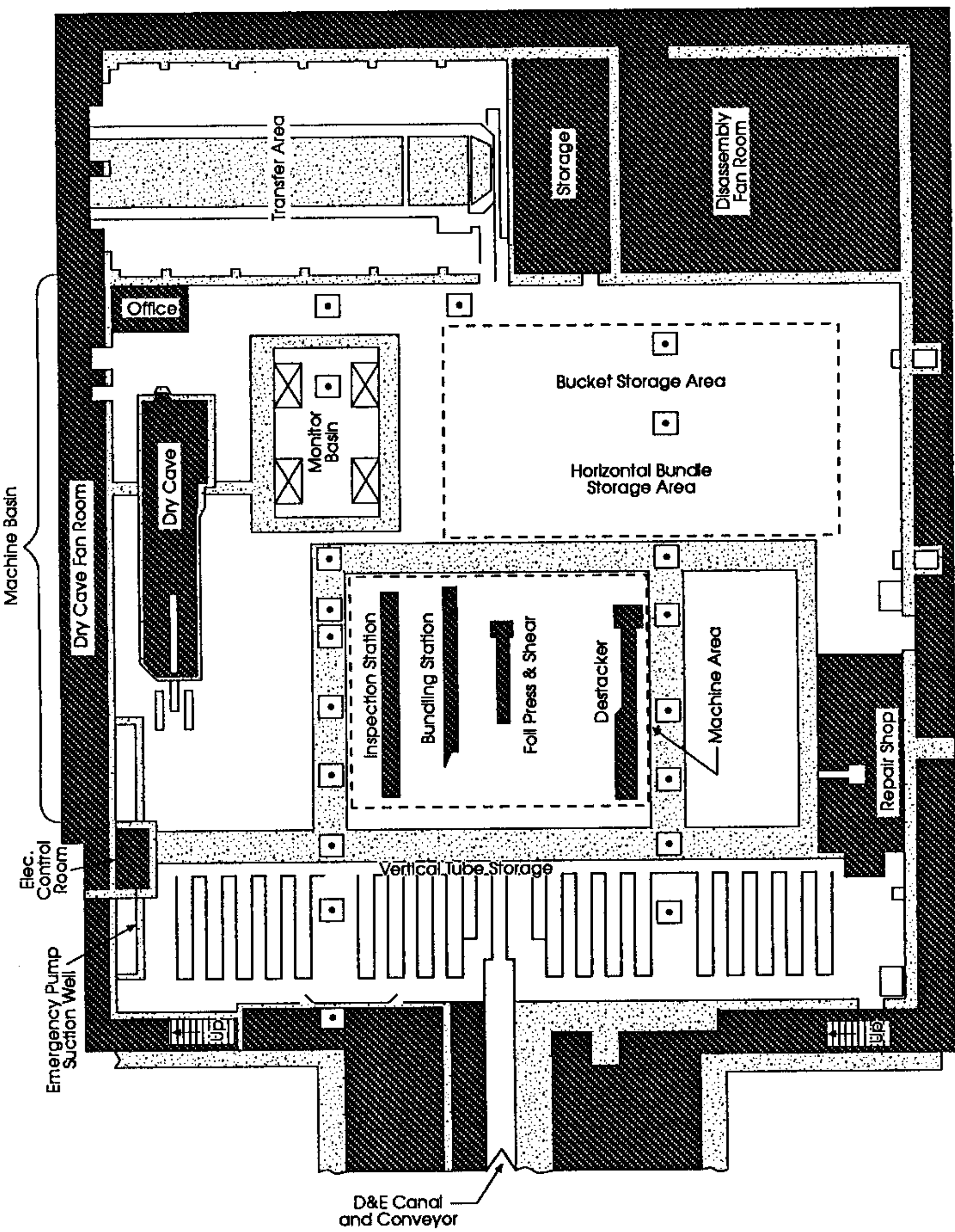


Figure F-41 Typical Reactor Disassembly Basin Area Layout

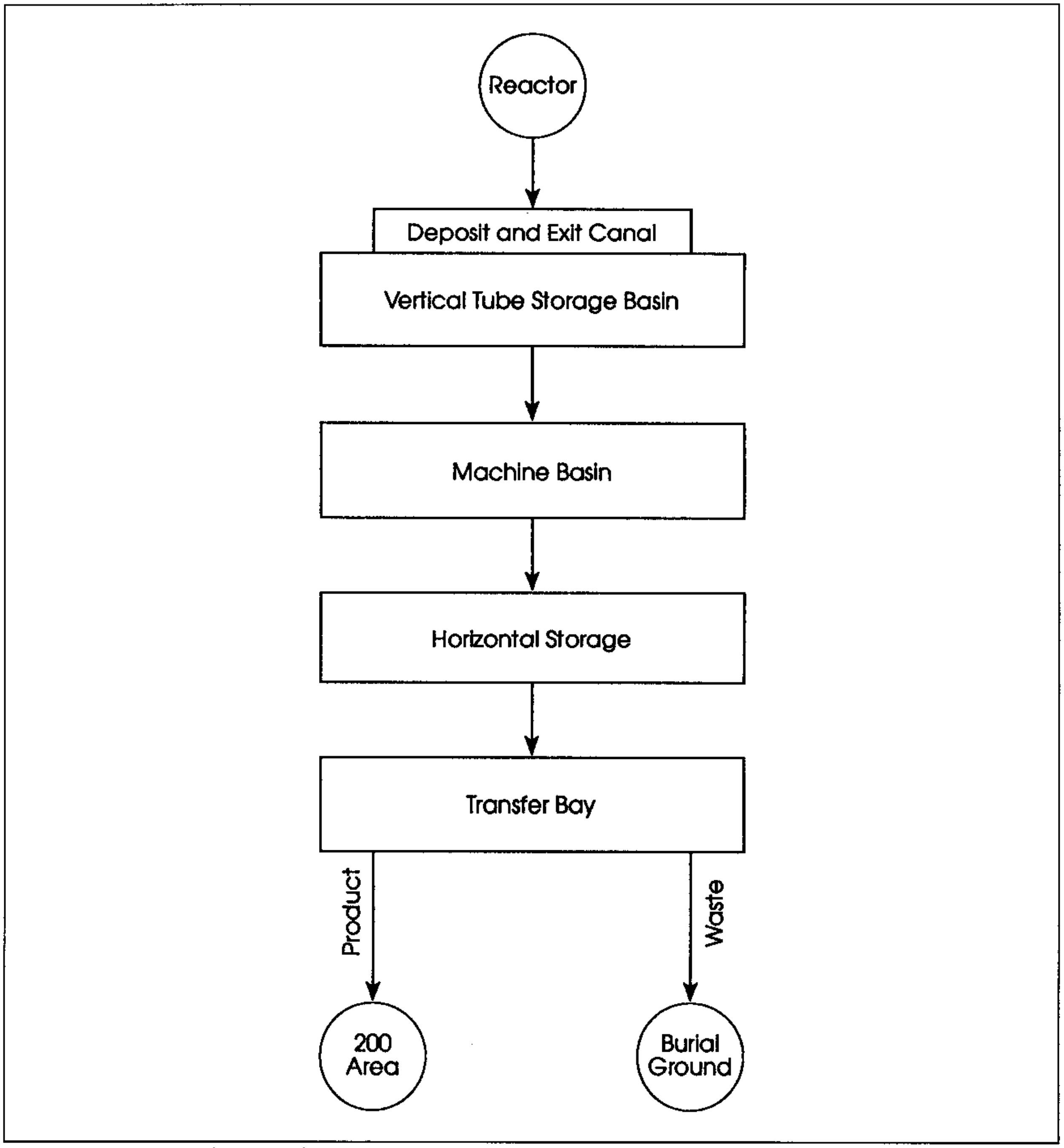


Figure F-42 Reactor Disassembly Basin Process Block Diagram

and transferred to the disassembly area side of the Deposit and Exit canal. The assemblies were transferred to hangers suspended from overhead monorails and were initially stored in the vertical tube storage area for 3 to 8 months. Fuel and target assemblies were moved to the machine basin area where they were disassembled. Target material was placed in stainless steel buckets and then stored in the bucket storage area. Fuel was bundled in aluminum bundles and stored in horizontal storage racks. The components were then allowed to cool for up to another 8 months. The fuel and target material was expected to be in the basin no longer than 18 months. The components, once sufficiently cooled, were moved to the transfer area to be shipped by cask to the processing facility.

Use of a disassembly basin for foreign research reactor spent nuclear fuel would require continuous demineralizer treatment for water quality, and new storage racks. Existing heat exchanger systems can remove upwards of 6,800 kilowatts (24 x 10^6 BTU/hr), which should far exceed the 240-1,000 kilowatts heat generation rate of foreign research reactor spent nuclear fuel (i.e., 10 to 40 Watts per element). These changes would allow each basin to accommodate approximately 20,000 elements.

The transfer area provides an area for shipping or receiving material and equipment. This area consists of two water-filled basins which are designated the scrap pit and the transfer pit. Irradiated material ready for shipping is transferred from horizontal storage to the transfer bay. Transportation casks are moved to and from the transfer pit and irradiated material placed in the cask using hoists mounted on the monorail system. Transportation casks can be transported to and from the reactor areas by tractor trailer or railroad. Trailers or railcars are positioned inside the transfer pit and casks are lifted and transported into/out of the basin using an 85/30 ton overhead crane.

Over the course of the site's history, at least 10 different casks have been used for various applications, many of which are still available for use pending proper inspection and maintenance. Two types are now used for most, if not all, disassembly work. EP-85 is a 63.5-metric ton (70-ton) fuel and target transport cask and EP-383 is a 13.6 metric ton (15-ton) cask used to move scrap to the burial ground.

The transfer area cranes would have to be modified to accommodate the different casks used for offsite shipments. These changes would allow a disassembly basin to receive up to seven casks per month in addition to the projected Savannah River Site shipping requirements.

A monorail system is mounted to the ceiling throughout the disassembly area. This system is used to transport and store all types of spent nuclear fuel and reactor components in the disassembly basin area. Most of the disassembly monorail system is designed for a working load of 907 kg (2,000 lbs) per foot of rail, which would be adequate for moving foreign research reactor spent nuclear fuel.

H-Canyon

The H-Area facilities occupy approximately 160 ha (395 acres). The H-Area Canyon processed irradiated fuel elements via modifications of the plutonium-uranium extraction process (Figure F-43) and is oriented toward HEU recovery. Primary operations also include the dissolution of fuel tubes, chemical and physical separations, and purification of materials. DOE stores the high-level waste from the operations in large tanks (nominally, 3.8 million l or 1 million gal each) for future stabilization and disposal via the Defense Waste Processing Facility.

The facility arrangement of the H-Canyon is identical to the F-Canyon. The main facility in the H-Area is the 221 H-Canyon, where most of the separations of irradiated materials were accomplished. The H-Canyon is a Class 1 reinforced concrete structure with exterior dimensions of 254.4 m (835 ft) by 37.2 m (122 ft) by 20.1 m (66 ft) high. The facility houses two parallel process canyons, each 9.1 m (30 ft) wide. The two process canyons handle high activity and lower activity materials, and are designated as the hot and warm canyons, respectively. Each of these process canyons is divided into 14 process cells, 13.1 m (43 ft) long by 4.6 m (15 ft) wide by 13.7 m (45 ft) high. Each process canyon is serviced by an overhead crane that operates the entire length of the canyon. The cranes perform remote operations for the processes such as equipment replacement, piping and electrical changes, leak repair, and inspections. Recently, new cranes were installed in both canyons. These cranes are remotely operated and include a Closed Circuit Television system for better monitoring capabilities. The warm process canyon crane has a lifting capacity of 27.2 metric tons (30 tons) and the hot process canyon's crane capacity is 45.4 metric

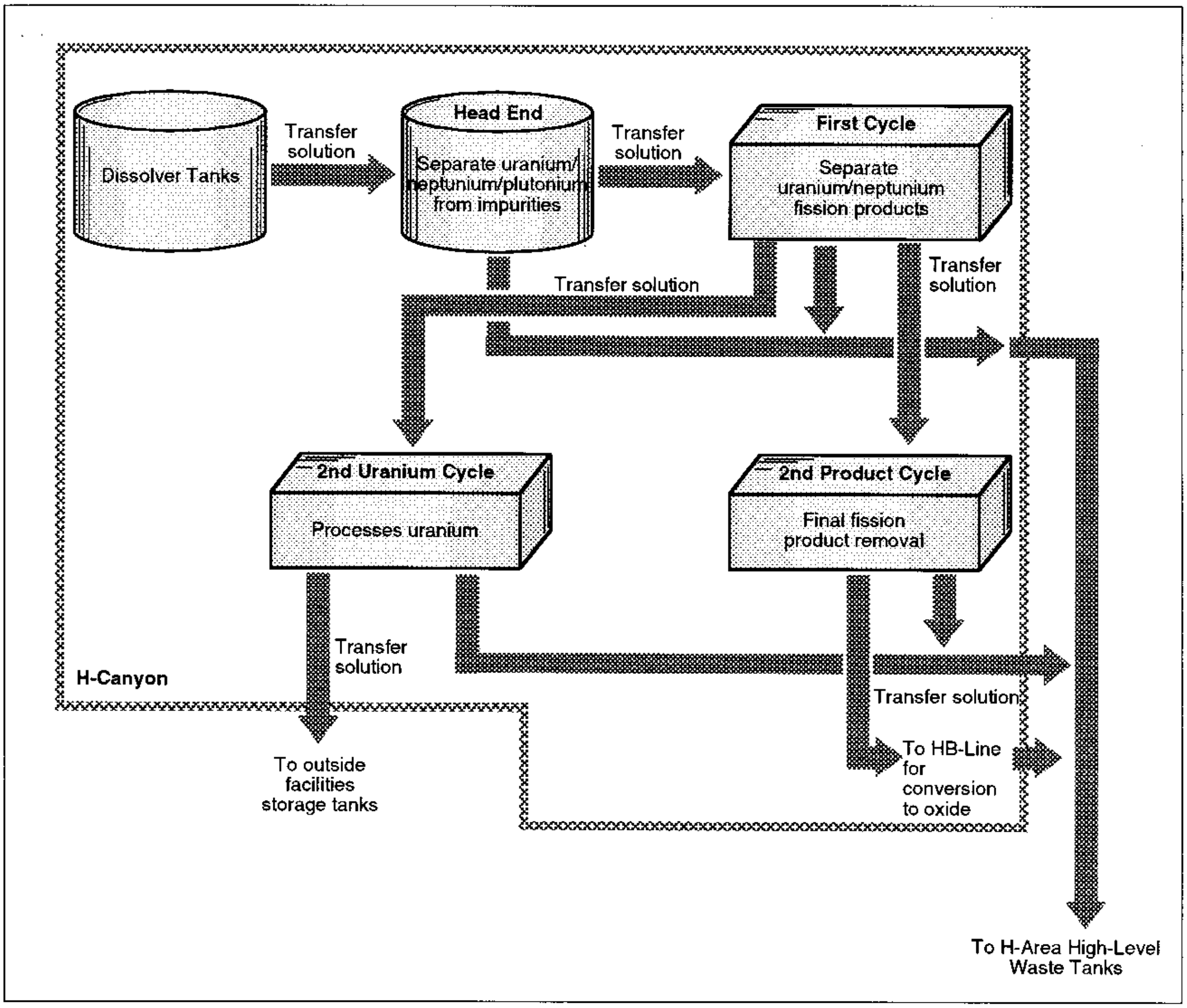


Figure F-43 H-Canyon Modified Plutonium-Uranium Extraction Process Flow

tons (50 tons). A separate maintenance area for each canyon provides for crane maintenance and repair. The original canyon cranes are still in place and can be utilized as backups to the new ones.

Spent nuclear fuel is delivered by a water-filled railway mounted cask into a shielded tunnel. The hot canyon overhead crane unloads spent nuclear fuel either into a storage basin in Section 3 or into the dissolver. The H-Canyon basin was never designed for storage of spent nuclear fuel, but only as a staging area prior to reprocessing. As such, it does not have the capability to establish, circulate, or maintain water chemistry. Basin water is sampled and analyzed every 3 months and domestic water is added manually every 3 months by opening an appropriate valve to the basin to make up for evaporation.

The H-Canyon storage basin is constructed of concrete and is 5.4 m (17.75 ft) long, 2.38 m (7.83 ft) wide, and 7.6 m (25 ft) deep. The floor is covered by a stainless steel liner that also extends 4.3 m (14 ft) up each wall. A stainless steel storage rack sits in the bottom of the basin and provides adequate spent nuclear fuel separation. The basin is filled with water to a level of 3.8 m (12.5 ft). The fuel currently being stored in the basin contains approximately 40 kg (88 lbs) of ^{235}U. The H-Canyon is currently in a status ready for restarting. An EIS was prepared to cover the canyon's restart for processing of the liquids currently stored in tanks within the facility. Future missions for the facility are still being analyzed. For foreign research reactor spent nuclear fuel, H-Canyon could be used for chemical separation and blending down of HEU to LEU material.

F.3.3.2 Idaho National Engineering Laboratory

On October 17, 1995, litigation with the State of Idaho was settled by stipulation of the parties and entry of a consent order. This settlement would provide for the transportation of up to 61 shipments of foreign research reactor spent nuclear fuel to the Idaho National Engineering Laboratory prior to the year 2000, if DOE and the Department of State choose to adopt a policy of accepting such spent nuclear fuel. After the year 2000, additional shipments of such spent nuclear fuel could be made to the Idaho National Engineering Laboratory under the stipulated settlement and consent order. However, the following discussion is to provide a full understanding of the existing capabilities at the Idaho National Engineering Laboratory in light of the fact that this site has been considered as a reasonable alternative to manage the foreign research reactor spent nuclear fuel as in the preparation of the Draft EIS.

Irradiated Fuel Storage Facility

The ICPP-603 includes an underwater fuel storage basin area and the IFSF, which is a remotely operated, dry-vault facility specifically constructed for the storage of graphite fuel from the Fort St. Vrain and Peach Bottom reactors. It was built in 1974 as an addition to the underwater Fuel Storage Facility and contains 636 storage positions. This facility can handle casks weighing up to 55 metric tons (60 tons). Spent nuclear fuel currently stored here is from two commercial high-temperature, gas-cooled reactors (Fort St. Vrain and Peach Bottom), some for the ROVER Nuclear Rocket Program, and some Tory 2C and BER II TRIGA fuel. The IFSF is a good candidate for spent nuclear fuel requiring frequent monitoring because of the ease of visual fuel inspections.

Since the facility can accommodate fuels up to 3.0 m (130 in) in length, all types of foreign research reactor spent nuclear fuel under consideration in this EIS could be handled. Transfer cart modifications would be needed for the proposed foreign research reactor spent nuclear fuel transportation casks, since the existing transfer cart only has the capability to handle the Rover cask, the Fort St. Vrain cask, and the Peach Bottom cask. New fuel handling tools, such as a new can grapple, would be needed. New cell preparations and work stations would also be needed. Sipping, unloading, canning, sealing, and leak checking equipment would need to be added to the cell. A 14 metric ton (15 ton) crane is present in the vault room for fuel handling. Visual inspection and gamma spectroscopy could readily be performed in the existing vault room.

The vault room is 7 m by 7.1 m by 6.6 m (22 ft 10 in by 23 ft 3 in by 21 ft 6 in) high, and the storage room is 437 m^2 (4,700 ft^2). Loaded fuel cans can be transferred from the vault room to the storage area by a shuttle bin. The vault room is being reanalyzed structurally to validate its capability to meet the current seismic requirements of 10 CFR 72. Recent reliable data regarding the effectiveness of the filtering and ventilation systems must be obtained in order to assess the amount of radionuclides that may be vented into the outside air. The cost to add the required capabilities to the IFSF for storage of foreign research reactor spent nuclear fuel is approximately $5 million.

It should be mentioned that, although this facility was originally constructed to accommodate the Fort St. Vrain High Temperature Gas-Cooled Reactor graphite fuel, it will not be used for this purpose because of the October 16, 1995 Settlement Agreement with the State of Idaho that declared that the Fort St. Vrain fuel will not be brought to the Idaho National Engineering Laboratory for interim storage. Public Service of Colorado has obtained a 10 CFR 72 license to store all of the fuel in the Foster Wheeler modular dry vault facility built adjacent to the reactor site. That modular dry vault is currently completely loaded with Fort St. Vrain fuel, and the reactor is being decommissioned. Availability of the IFSF for foreign research reactor spent nuclear fuel will be dependent on decisions to consolidate spent nuclear fuel from other Idaho National Engineering Laboratory facilities at this facility.

Approximately 300 positions are available in the IFSF dry storage facility for foreign research reactor spent nuclear fuel. Preparations to receive the foreign research reactor spent nuclear fuel could be completed as soon as calendar year 1997. However, many activities are already scheduled for this facility. A new canning station for spent nuclear fuel from the ICPP-603 basins is being constructed in the handling cave, and the canning will then be accomplished. Other spent nuclear fuel management activities being considered in this facility include ROVER reactor fuel shipments, Experimental Breeder Reactor II and FERMI movements fuel from ICPP-666. Naval fuel inspection sample receipts from the Expended Core Facility are scheduled. The Peach Bottom fuel in the ICPP-749 facility and Rover fuel ash transfers from a shutdown fuel processing facility are also being considered for repackaging in the IFSF handling cell, and the Idaho National Engineering Laboratory spent fuel consolidation activities are scheduled to begin within 2 years. Detailed facility usage schedules have been drafted to demonstrate how the foreign research reactor spent nuclear fuels could be accommodated in this workload.

With this schedule as background, foreign research reactor spent nuclear fuel preparations could take place during early 1996. Fuel could be received beginning in 1997 (61 shipments through FY2000) and up to 101 shipments in the years beyond this timeframe.

The 300 positions could store up to approximately 9,000 foreign research reactor spent nuclear fuel elements. Approximately 60 dry storage positions in the ICPP-749 drywells could also be utilized to store foreign research reactor spent nuclear fuel. Some refurbishment would have to take place to receive spent nuclear fuel. This could be completed in 1997, and the facility would be ready to receive fuel at the beginning of 1998. With 60 fuel elements per position, up to 3,600 fuel elements could be stored in the ICPP-749. Any fuel that is stored in the ICPP-749 would have to go through the ICPP-603-IFSF to be placed in sealed containers and transferred to an interfacility transfer cask.

Idaho Chemical Processing Plant-666 Fuel Storage Area

This facility (Figure F-44) is the modern Idaho National Engineering Laboratory underwater storage facility. Receipt and storage of foreign research reactor spent nuclear fuel has been accomplished in the past as one of its many missions. It has the capability of receiving and unloading spent nuclear fuel casks at a rate of approximately five per week. Storage capability for up to 8,400 foreign research reactor spent nuclear fuel elements can be provided for an approximate 10-year period by using the increased capacity fuel storage racks that will be installed in Pool 1 via a reracking project planned following the Programmatic SNF&INEL Final EIS, and by installing additional fuel storage racks in the cutting pool. The increased capacity being provided in Pool 1 will be required for Naval spent nuclear fuel receipts beginning in about FY 2005. The racks being removed from Pool 1 as part of the reracking project could be placed in the cutting pool for the foreign research reactor spent nuclear fuel.

The capability of the ICPP-666 facility to receive foreign research reactor spent nuclear fuel in the near term is limited, however, due to the number of activities scheduled through 1998. These activities include reracking in Pools 1, 6, and 5. Fuel receipts from the Navy and the Advanced Test Reactor, and fuel transfers from CPP-603 will continue. The CPP-603 fuel transfers are required to meet a court order requirement. These activities utilize nearly all resources, such as cranes, manpower, and health physics personnel in the near term of 1995 through 1997. A limited number of shipments (one per month) are possible for the 1996 and 1997 schedules. In 1998, the schedule relaxes enough that up to 30 shipments

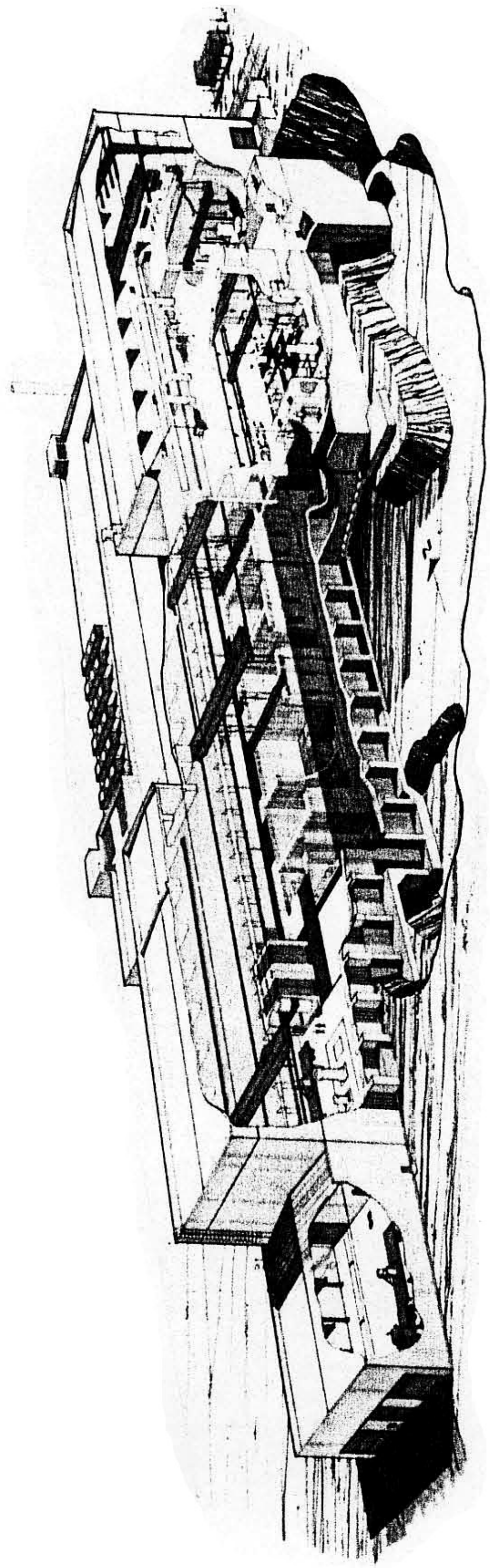


Figure F-44 Pictorial of the Idaho Chemical Processing Plant-666

could be received because most of the ICPP-603 fuel transfers to Fluorinel Dissolution and Fuel Storage (FAST) will have been completed. By the end of 1999, a total of 3,600 fuel elements could be received under this scenario. This schedule is predicated on the assumption that resolution of Idaho National Engineering Laboratory facility vulnerabilities and Naval and Advanced Test Reactor fuel receipts have a higher priority than foreign fuel receipts.

Idaho Chemical Processing Plant-666 Fluorinel Dissolution Process

The conversion of the ICPP-666 Flourinel Dissolution Process cell for canning of spent nuclear fuel without removal of any existing equipment or decontamination of the cell is proposed as a low-cost option to prepare foreign research reactor spent nuclear fuel for dry storage. Only minor modifications would be made to the cell, so as to preserve the current dissolution capability for possible future use. This option would utilize the Flourinel Dissolution Process cell, which currently has remote fuel handling, sampling, and waste load-out capabilities, as well as a connection to the ICPP-666 Fuel Storage Area, for fuel inspection, stabilization, and packaging for interim dry storage.

A potential disadvantage is noted. The equipment inside the Flourinel Dissolution Process cell is contaminated, and radiation fields are too high for manned entry. Retaining the dissolution cell equipment will make it impossible to adequately clean the cell to allow personnel entry. For this reason, equipment requiring installation within the Flourinel Dissolution Process cell must be assembled outside the cell and installed remotely with the in-cell crane and master-slave manipulators. Preventative and corrective maintenance of the equipment inside the cell would be done remotely. The modular design of the components would facilitate removal and replacement.

No general Flourinel Dissolution Process utility upgrades, such as electrical power or ventilation, would be required. Piping services could be added to support the vacuum drying and inert gas backfilling functions envisioned to meet dry storage requirements. All of the Reduced Enrichment for Research and Test Reactors (RERTR) program spent nuclear fuel could be accommodated by the existing transfer tunnel and transfer cart. Modifications of the existing fuel shear tools, or new ones, could be acquired to shear larger objects such as cans and lids.

Fuel Processing Restoration

Another structure that may represent an option for fuel storage is the Fuel Processing Restoration building (Figure F-45) that was constructed to house the Fuel Processing Restoration process. It is approximately 56.4 m (185 ft) long and contains shielded, below-grade process cells. These cells vary in dimension, with the nine main process cells measuring 5 to 6 m (16.5 to 20 ft) wide, 10.4 m (34 ft) long, and 12.2 m (40 ft) deep. Fuel racks could be designed to accommodate cans of the type proposed for dry storage in arrays that could contain as many as 17 cans along the 10.4 m (34 ft) axis by 8 cans along the 4.9 m (16 ft) axis, and be 3 cans deep. Airflow through each of these cells could be controlled by dampers in the cell ductwork. Construction of this facility was interrupted prior to completion, and it currently does not include cell ventilation, fire safety equipment, instrumentation, or lighting. These additions would cost approximately $15 million. Several other processes are being considered for use of this facility, and there is no assurance that it would be used for fuel storage. When completed, the building will be seismically qualified, and could physically accommodate approximately 540 canisters per main process cell. Special remote handling tools and techniques would be developed to allow the fuel cans to be inserted into the storage cell. The total estimate to make all necessary conversions is $65 million.

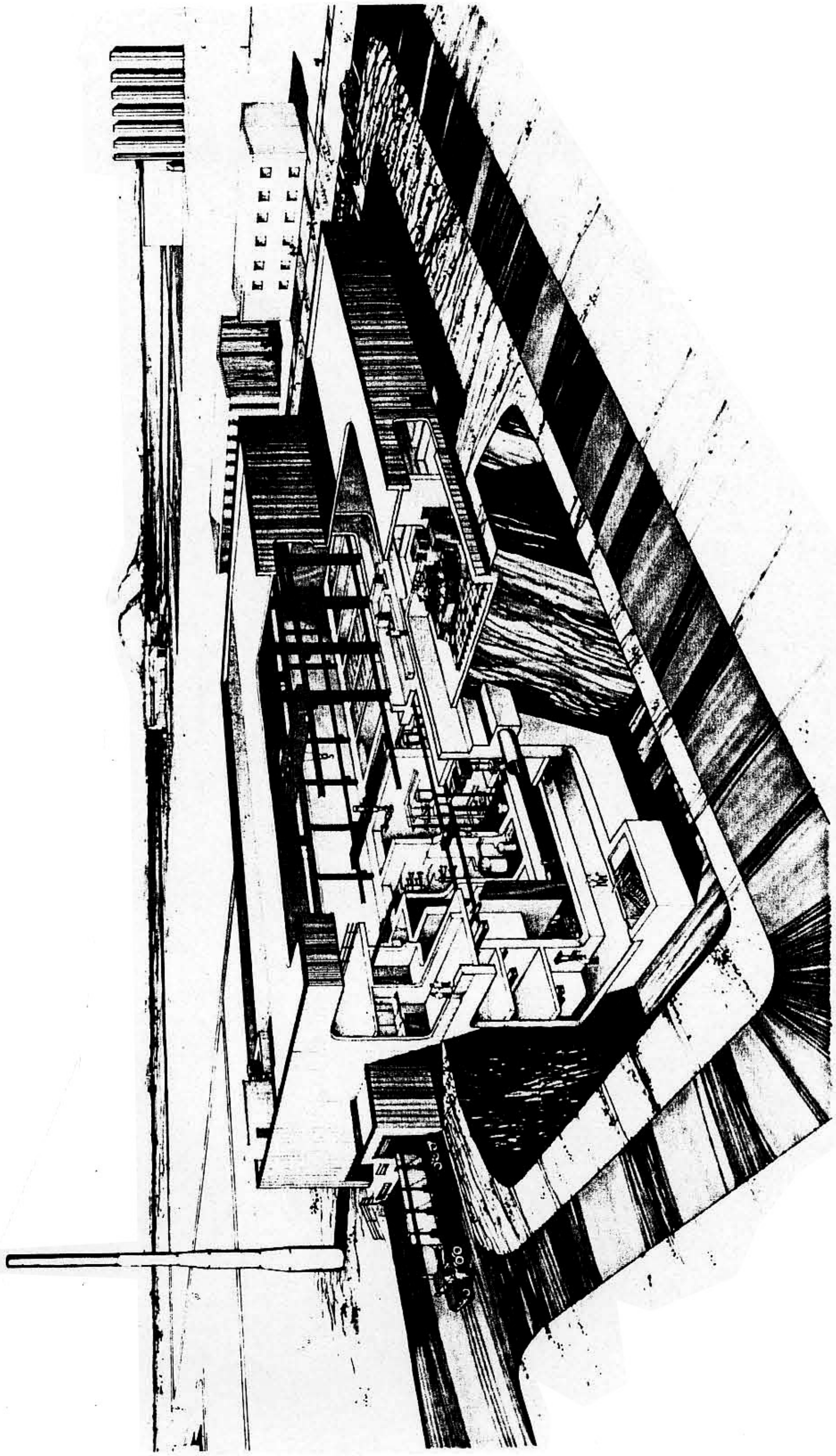


Figure F-45 Fuel Processing Restoration Facility (Unfinished)

Hot Fuel Examination Facility

The Hot Fuel Examination Facility is a facility used for examining and storing irradiated fuels from the EBR-II breeder reactor at Idaho National Engineering Laboratory. Figure F-46 presents the layout of the main cell of the Hot Fuel Examination Facility. Although it was not designed for storage, the Hot Fuel Examination Facility could be used to receive, inspect, examine, and transfer foreign research reactor spent nuclear fuel to dry cask storage if it is fitted with an appropriate spent nuclear fuel examination station. The cost of these modifications and the purchase and installation of the dry casks and their equipment would be the principal costs involved.

Test Area North-607 Pool, Hot Cell, and Cask Storage Pad

The utilization of the Test Area North-607 facilities is a potential option for receipt and storage of the foreign research reactor spent nuclear fuel. This could be accomplished without significant modification to the hot cell. The hot cell has significant lag capacity for interim storage of foreign research reactor spent nuclear fuel and has most of the equipment necessary for placement of the fuel into dry interim storage casks. There is adequate space for installation of the characterization and conditioning equipment needed for dry storage. There are significant vulnerabilities associated with the underwater storage pool which would need to be corrected if underwater interim storage were desired. The cask storage pad could be easily expanded to accept additional dry storage casks. At the current time, the entire Test Area North area is being planned for shutdown in approximately ten years due to reduced mission needs. If Test Area North had adequate new missions and it was determined to be economical, the Test Area North hot cell and cask storage area would have significant capacity for receipt and temporary storage of foreign research reactor spent nuclear fuel.

F.3.3.3 Hanford Site

In addition to the generic dry and wet storage facilities, two existing facilities at Hanford Site have been identified as potential candidates for the storage of foreign research reactor spent nuclear fuel: the FMEF and the Washington Nuclear Plant-4 Spray Pond. They will be discussed in greater detail in the following sections.

F.3.3.3.1 Fuel Maintenance and Examination Facility (FMEF)

The FMEF, built during the late 1970s and early 1980s (but never completed), consists of a 82.3 m (270 ft) long, 53.3 m (175 ft) wide, 29.9 m (98 ft) high Process Building with attached mechanical equipment and entry wings. The FMEF was intended to receive and extensively examine irradiated breeder reactor test fuels. The seismically-qualified FMEF Process Building, which extends 10.7 m (35 ft) below the surface, consists of 6 operating floors or levels and encloses a total of 17,466 m^2 (188,000 ft^2) of operations space. The FMEF has a 68 metric ton (75 ton) overhead crane, 18 metric ton (20 ton) hoist, and 9 metric ton (10 ton) hoist. Three areas within the FMEF, the Shipping and Receiving area, Decon Cell, and the Entry Tunnel would be used for the storage of foreign research reactor spent nuclear fuel. Figure F-47 presents a ground floor plan, and Figure F-48 shows vertical storage of spent nuclear fuel canisters.

The Shipping and Receiving Area, also known as Room 300, is the access area to the FMEF for truck or rail shipments of spent nuclear fuel. This area provides a 24.4 m (80 ft) long working area, washdown and decontamination of casks and shipping vehicles, and crane interface access to other areas within the FMEF. The 68 metric ton (75 ton) crane and hoists can transport casks to the Entry Tunnel from this area. The principal modification needed for this area would be the construction of a 0.8 km (0.5 mi) rail extension to the existing Hanford Site rail system.

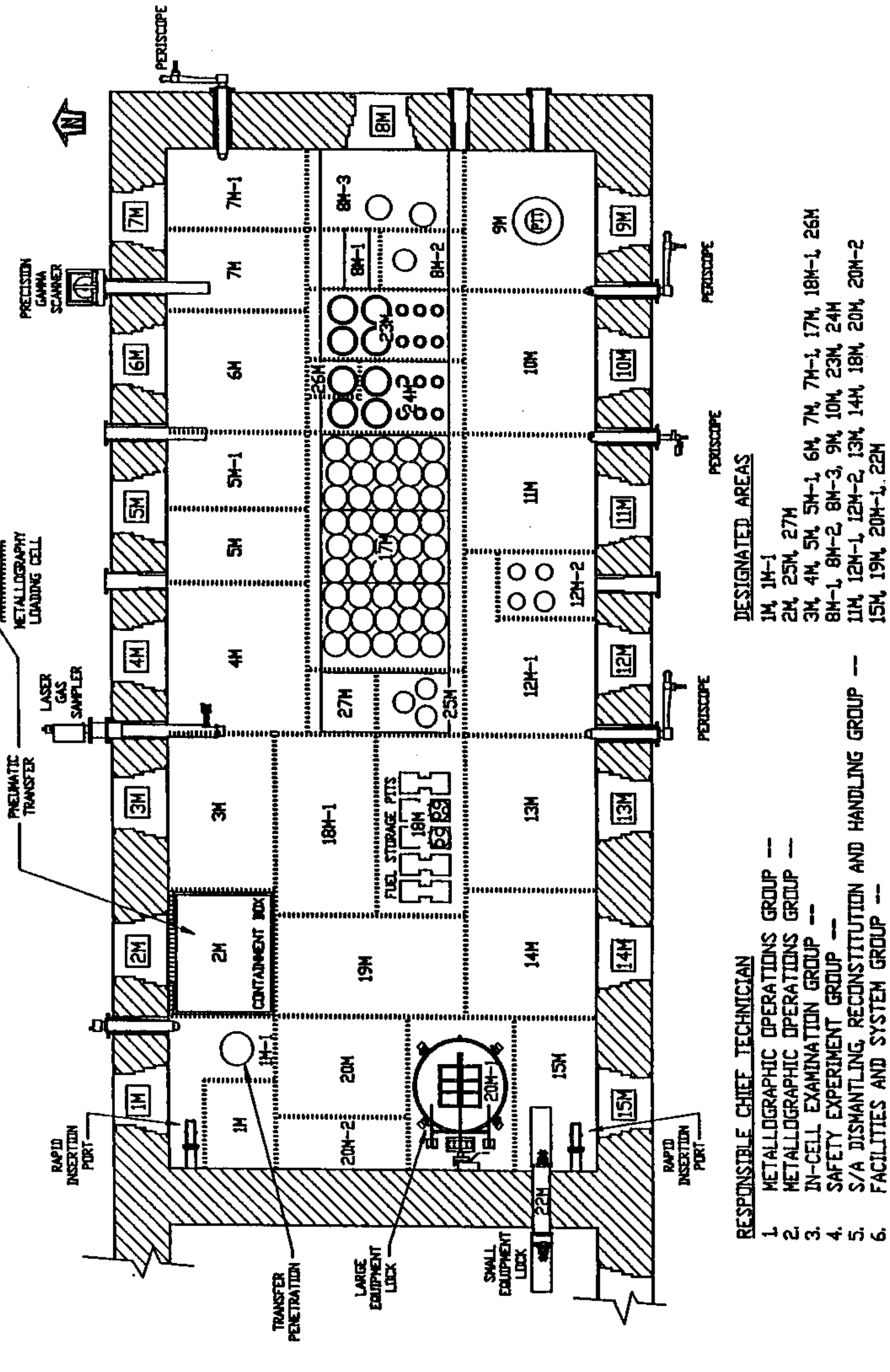


Figure F-46 Hot Fuel Examination Facility Main Cell Layout

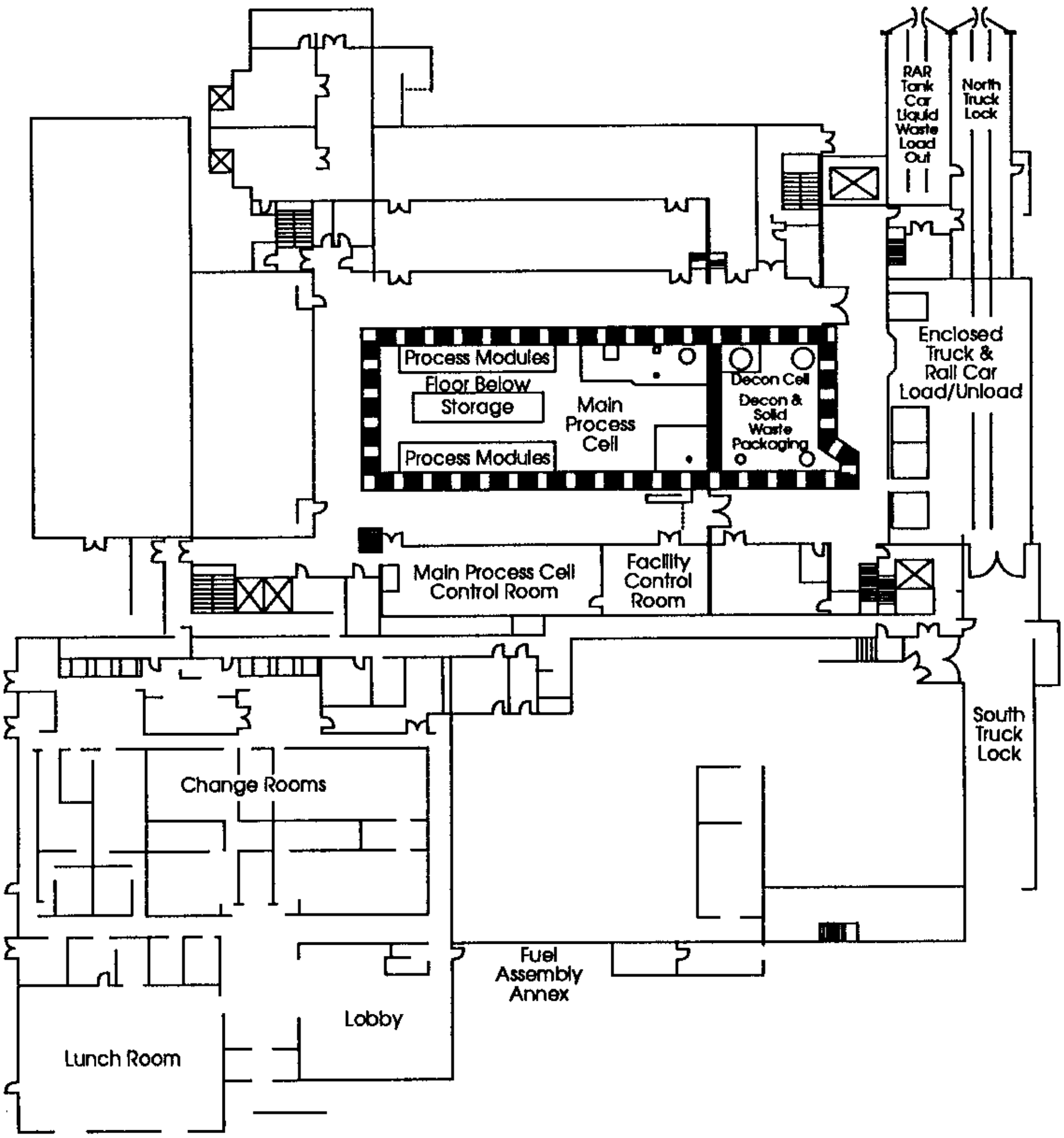


Figure F-47 Fuel Maintenance and Examination Facility 0'-0" Level Floor Plan

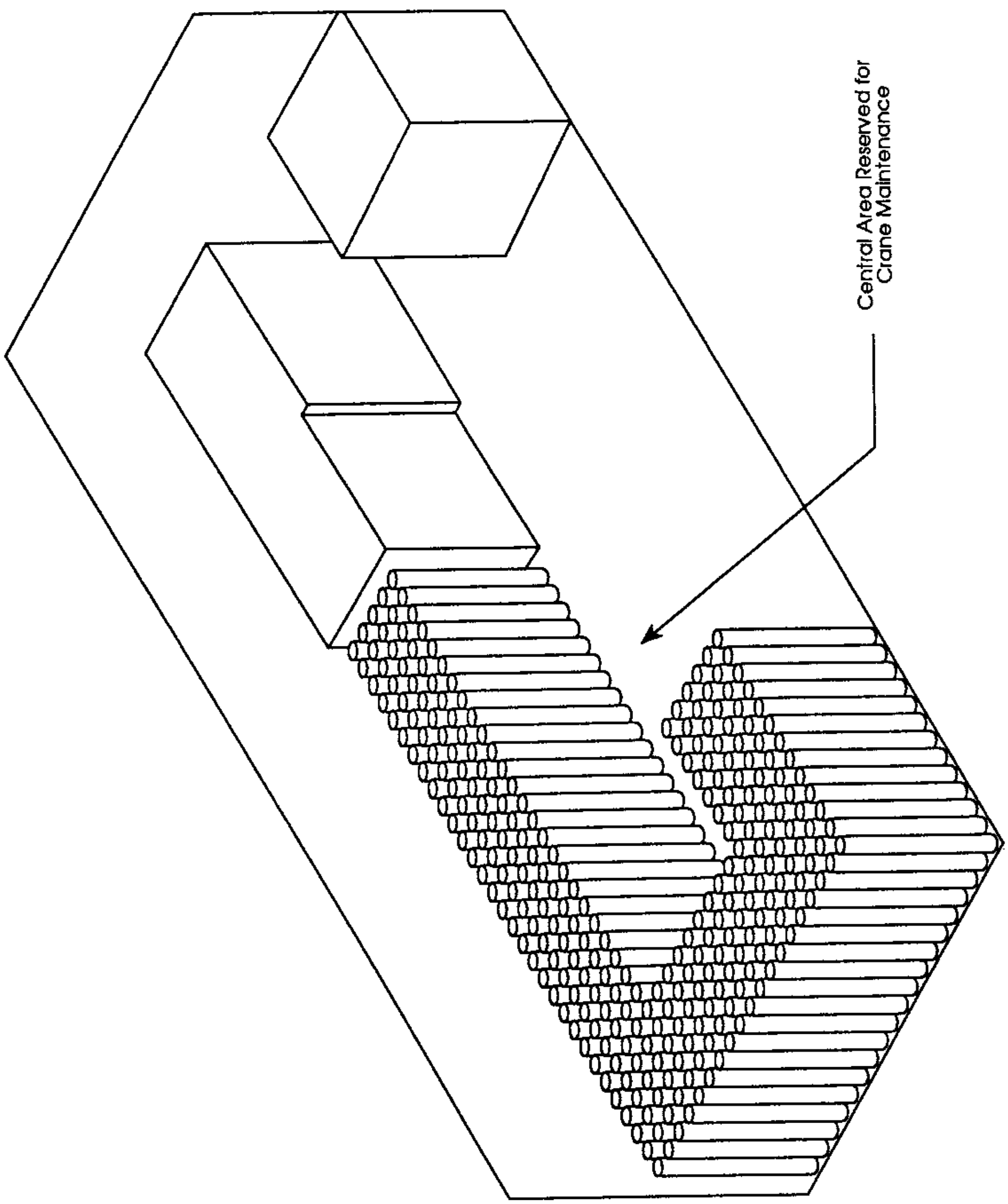


Figure F-48 Potential Use of the Fuel Maintenance and Examination Facility for Fuel Storage Partial Plan View

The Entry Tunnel is a 9.9 m (32.5 ft) high tunnel below the Shipping and Receiving area floor designed to transfer transportation casks to the Decon and Main Process Cell areas of the FMEF. It includes a 68 metric ton (75 ton) overhead bridge crane. To transport heavier multi-purpose casks when the foreign research reactor spent nuclear fuel would be shipped out of the FMEF in the future, the tunnel would be extended and modified to accommodate an above-grade 114 metric ton (125 ton) crane. The entry tunnel would be extended to connect to the new adjacent storage facility.

The Decon Cell is a room 12.2 m (40 ft) long, 9.1 m (30 ft) wide, and 11.6 m (38 ft) high, with thick concrete shielding. Nine work stations with remote manipulators and viewing windows are part of the design of this cell, although it should be noted that the viewing windows and equipment have not been installed. Access is available through two 2.1 m (84 in) diameter hatches, a 0.8 m (30 in) diameter port, and a 0.3 m (12 in) diameter opening. The Decon Cell includes material handling capability by cranes, manipulators, and hoists ranging from 1.4 to 6.8 metric tons (1.5 tons to 7.5 tons). The Decon Cell would be used to inspect foreign research reactor spent nuclear fuel that has been unloaded from transportation casks and to subsequently load this spent nuclear fuel into storage baskets. Each basket holds three foreign research reactor spent nuclear fuel elements. After basket loading, four baskets would be stacked into a 4.6 m (15 ft) high stainless steel canister. The canisters would be moved to the adjacent storage facility using a Transfer Tunnel, which is equipped with a cart.

The Main Process Cell represents a potential storage location for foreign research reactor spent nuclear fuel at the FMEF (Figure F-47). This room is 30.5 m (100 ft) long, 12.2 m (40 ft) wide, and 11.6 m (38 ft) high with concrete walls either 1.2 or 1.5 m (4 or 5 ft) thick, depending on the concrete's density. The Main Process Cell design includes two 4.5 metric ton (5 ton) bridge cranes and two 1.4 metric ton (1.5 ton) electro-mechanical manipulators.

A zoned heating, ventilation, and air conditioning system with negative differential pressure, redundant cooling systems, and staged multiple High Efficiency Particulate Air filters provides decay heat removal and protection from environmental releases of radioisotopes. This system provides for flow, by negative air pressure differential, from the least contaminated zones to the most contaminated zones, thereby maintaining individual zone relative contamination potential. Supply air is drawn from tornado-hardened and seismically qualified intake shafts and dampers. All heating, ventilation, and air conditioning equipment required to supply high contamination zones is designed as Seismic Category 1. After multiple High Efficiency Particulate Air filtration and monitoring for radioactivity, heating, ventilation, and air conditioning exhaust air is released from a seismically qualified reinforced concrete 35.7 m (117 ft) tall stack.

The FMEF is provided normal power by two separate 115 kV electric power supply lines from the Bonneville Power Administration. Emergency power is provided by two 100 percent redundant 900-kilowatt gas turbines which, along with their seismically qualified support and fuel oil systems, are capable of 24 hours of continuous operation. An Uninterruptible Power Supply, consisting of two 150-kVA lead calcium batteries, can provide full load for 30 minutes. Emergency generators require 2 minutes to start up and produce rated power.

A number of modifications would be required for the FMEF to be used as a storage facility for foreign research reactor spent nuclear fuel. They can be categorized as: addition of a 114 metric ton (125 ton) crane, railroad tunnel extension, and storage rack canisters. Even with these modifications, the FMEF does not have sufficient space to store 23,000 foreign research reactor spent nuclear fuel elements, but it could be used as an unloading and support facility for an adjacent dry vault storage facility. Costs for the necessary modifications to the FMEF have been estimated to be approximately $32 million. The adjacent dry storage facility is estimated to cost an additional $100 million. It should be noted that the FMEF is being considered for other spent nuclear fuel storage which could eliminate it for use with foreign research reactor spent nuclear fuel.

F.3.3.3.2 Washington Nuclear Plant-4 Spray Pond Wet Storage

The Washington Nuclear Plant-4 Spray Pond is a nuclear safety-related structure that was originally designed for decay heat removal following a Loss of Coolant Accident at the Washington Nuclear Plant-4 commercial nuclear power plant. The Washington Nuclear Plant-4 was canceled, but the spray pond structure is essentially complete. This pond is 91.4 m (300 ft) long, 76.2 m (250 ft) wide, and 8.2 m (27 ft) deep, and was designed and built to 10 CFR 50 Appendix B quality assurance standards as a seismic and safety class structure. It should be noted that the size of this spray pond is much greater than that needed to store the foreign research reactor spent nuclear fuel. Figure F-49 presents a schematic of the Washington Nuclear Plant-4 Spray Ponds and the necessary modifications required to store foreign research reactor spent nuclear fuel.

In order for this pond to be used for wet storage of foreign research reactor spent nuclear fuel, several modifications would have to be made to duplicate the features of the Generic Pool Facility. These modifications include: an enclosure with a qualified building superstructure, inclusion of a shipping-receiving-handling facility in this structure; a deeper loading pool; a 114 metric ton (125 ton) loading crane; a 274 m (900 ft) long railroad line extension; installation of heating, ventilation, and air conditioning, and pond water cooling and water chemistry/cleaning systems; canister support racks into the pond; installation of a stainless steel liner in the pond; installation of partition walls to isolate the fuel storage and handling areas from the unused portion of the spray pond; and a leak detection system. The supporting pumphouse is a 12 m (40 ft) by 30 m (100 ft) concrete building designed to the same standards as the spray pond and would be used to house some of the fuel handling and storage support equipment. Total cost for these modifications was estimated to be approximately $113 million (Bergsman et al., 1994).

F.3.3.4 Oak Ridge Reservation

No existing facilities at the Oak Ridge Reservation are being used for the storage of foreign research reactor spent nuclear fuel. However, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Oak Ridge Reservation.

F.3.3.5 Nevada Test Site

No existing facilities at the Nevada Test Site are being used for storage of foreign research reactor spent nuclear fuel, although the Area 25 facilities (E-MAD/Reactor Maintenance and Disassembly) have been used in the past and might be suitable in 1 to 3 years. These Area-25 facilities appear capable of accommodating the required cask receipt rate and dry storing all of the foreign research reactor spent nuclear fuel under consideration in this EIS. However, in addition to Area 25, either the generic wet (pool) or dry storage facilities, as described in Sections F.3.1 and F.3.2, could be constructed and operated at Nevada Test Site in Area 5.

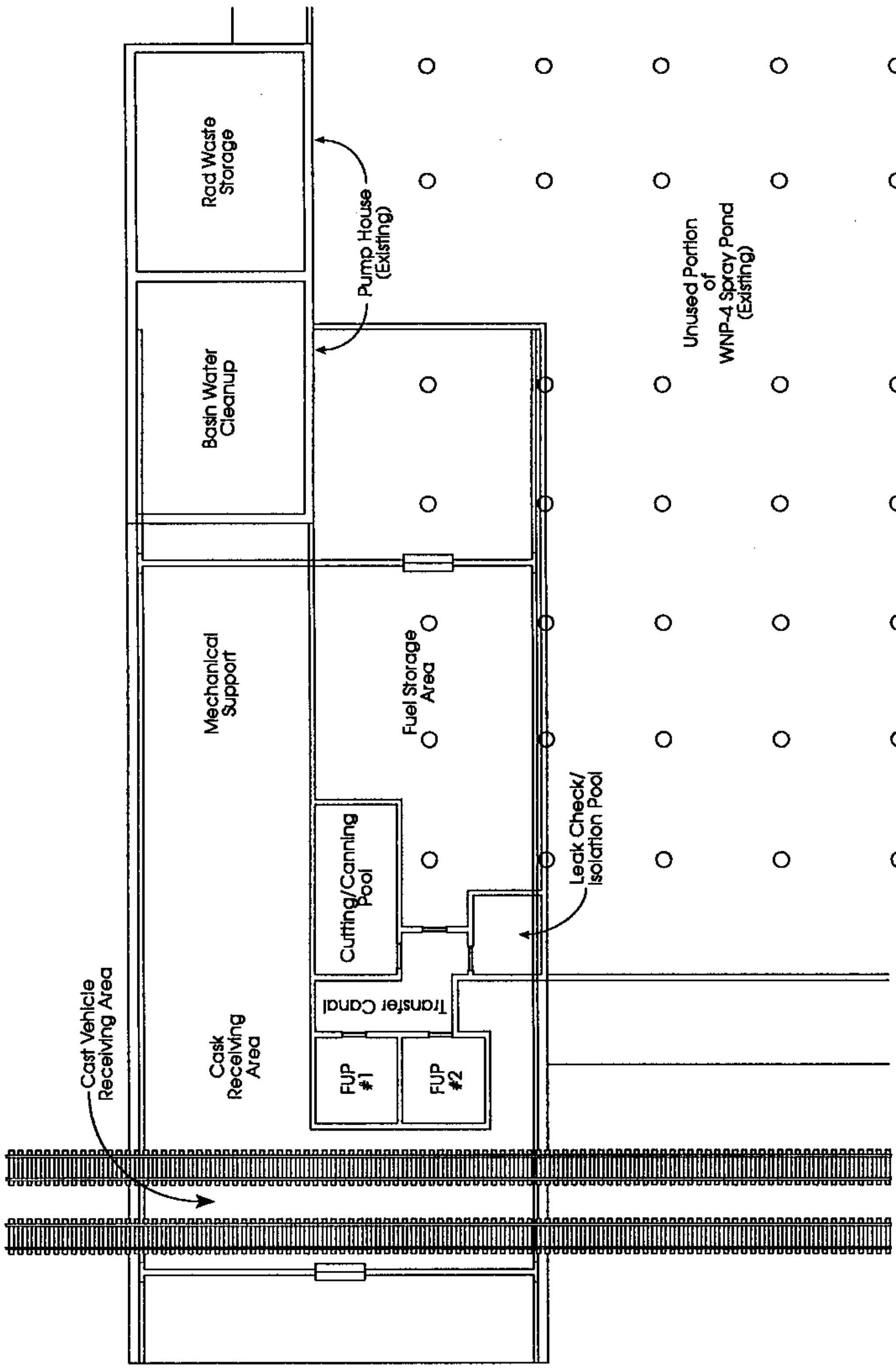


Figure F-49 Washington Nuclear Plant-4 Spray Pond (with Modifications) Schematic

F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

This section analyzes the environmental impacts associated with the storage of foreign research reactor spent nuclear fuel at the five potential management sites considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g), namely: the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site.

The Record of Decision for the Programmatic SNF&INEL Final EIS was issued on May 30, 1995. In accordance with this Record of Decision, all of the aluminum-based foreign research reactor spent nuclear fuel accepted by DOE would be managed at the Savannah River Site in South Carolina, and any other foreign research reactor spent nuclear fuel to be accepted by DOE would be managed at the Idaho National Engineering Laboratory. Nevertheless, all five of the spent nuclear fuel management sites originally considered in this EIS and the spent nuclear fuel distribution alternatives have been kept in the final to maintain maximum consistency with the analyses provided in the Programmatic SNF&INEL EIS (DOE, 1994h and 1995g).

The environmental impacts analyzed pertain to management of foreign research reactor spent nuclear fuel under the basic implementation, and implementation alternatives of Management Alternative 1, the storage of vitrified waste that may be accepted by the United States under Management Alternative 2, and the management of foreign research reactor spent nuclear fuel under Management Alternative 3, the Hybrid Alternative. Chemical separation, which is an implementation alternative to storage, is analyzed in Section 4.3 of this EIS.

Since foreign research reactor spent nuclear fuel is part of the DOE's overall management of spent nuclear fuel, the management options in this EIS must be consistent with the site management alternatives considered in the Programmatic SNF&INEL Final EIS. The alternatives considered in the Programmatic SNF&INEL Final EIS are: Decentralization and 1992/1993 Planning Basis (even distribution of foreign research reactor spent nuclear fuel between the Idaho National Engineering Laboratory and the Savannah River Site), Regionalization by Geography, Regionalization by Fuel Type, and Centralization (all foreign research reactor spent nuclear fuel eligible under the policy).

The foreign research reactor spent nuclear fuel management options also depend on the availability of the sites to implement the policy immediately. Of the five sites, only the Savannah River Site and the Idaho National Engineering Laboratory would be available in late 1995. The other three sites could become available at a later date when appropriate facilities would be completed (either constructed or refurbished). This constraint has necessitated a two-phased approach to foreign research reactor spent nuclear fuel management in the United States in which foreign research reactor spent nuclear fuel is received and managed first at an available management site and is shipped to another site later. For the purpose of this analysis, the implementation of the policy was divided into two functional periods — the period during which receipt and storage of foreign research reactor spent nuclear fuel would be accomplished by using existing facilities (Phase 1), and the period during which new or refurbished facilities could be used (Phase 2). The first phase would be characterized by operational activities only, while the second involves impacts from construction in addition to operational activities.

The environmental impacts from the basic implementation of each Management Alternative, as they relate to storage of the foreign research reactor spent nuclear fuel in the United States, are analyzed in Sections F.4.1 through F.4.5. Elements of this analysis are combined and summarized in Section 4 of this EIS to present the impacts of all Implementation Alternatives under the proposed action.

F.4.1 Savannah River Site

If the Savannah River Site is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate disposition. If the Savannah River Site is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Savannah River Site until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years. Modifications to existing facilities for the same purpose could take less time. This period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2. The amount of spent nuclear fuel that could be received and managed at the Savannah River Site under Management Alternative 1, as discussed in Section 2.2.2, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Savannah River Site could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, the aluminum-based foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, the foreign research reactor spent nuclear fuel from eastern ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site under Management Alternative 1, the Savannah River Site would receive and manage foreign research reactor spent nuclear fuel at existing wet storage facilities: RBOF and the L-Reactor disassembly basin. Descriptions of RBOF and the L-Reactor disassembly basin are provided in Section F.3. RBOF is located at the H-Area. It is a facility with provisions for the receipt and storage of irradiated nuclear fuel elements. Since 1963, irradiated spent nuclear fuel elements have been received from offsite reactors and from the Savannah River Site reactors. RBOF provides the capability for underwater unloading of the transportation casks and the handling and storage of the foreign research reactor spent nuclear fuel. The foreign research reactor spent nuclear fuel would be stored in RBOF until its storage capacity is exhausted. Currently, RBOF has space for approximately 1,170 foreign research reactor spent nuclear fuel elements. This capacity could be increased to a total of 2,425 elements by rearrangement and consolidation of existing inventory (O'Rear, 1995).

The L-Reactor disassembly basin is not currently configured for storage of aluminum-based foreign research reactor spent nuclear fuel; however, minor modifications which would provide new storage racks, new handling equipment, safety documentation, etc., along with upgrades in progress to address vulnerabilities associated with water chemistry control, would permit receipt and management of foreign research reactor spent nuclear fuel. Installation of racks equivalent to those in RBOF would provide storage for approximately 20,000 foreign research reactor spent nuclear fuel elements. The modifications to RBOF and L-Reactor disassembly basin are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between the RBOF and the L-Reactor disassembly basin, there would be sufficient storage capacity and handling capability to accommodate the receipt and management of foreign research reactor spent nuclear fuel during the estimated 10-year period for Phase 1.

An additional option to enhance storage capacity during Phase 1 would be to use RBOF and/or L-Reactor disassembly basin to unload the transportation casks and provide storage capacity in dry storage casks which would be placed near the existing facility. Descriptions of the dry storage casks are provided in Section F.3.

As a Phase 2 site under the basic implementation of Management Alternative 1, the Savannah River Site would continue to receive foreign research reactor spent nuclear fuel beyond Phase 1 in a new dry storage facility that would be constructed at the H-Area. The H-Area is the preferred site among several considered for the construction of new foreign research spent nuclear fuel storage facilities, and is the location assumed for the environmental impacts calculations. An alternative site, equally qualified for construction of new storage facilities is located on a ridge between the P-Reactor and the Pen Branch watershed as indicated in Section 2, Figure 2-14 of this EIS (Shedrow, 1994a). Foreign research reactor spent nuclear fuel stored during Phase 1 would be transferred to the new facility and would be stored there for an additional 30 years until ultimate disposition. The dry storage would encompass a number of designs, examples of which were provided in Section 2.6.5.1.1 and in Section F.3.

The analysis of environmental impacts from the management of foreign research reactor spent nuclear fuel at the Savannah River Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

1A. The Savannah River Site would receive foreign research reactor spent nuclear fuel during Phase 1 and store it at the RBOF and/or the L-Reactor disassembly basin. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received during Phase 1 (approximately 17,500 elements). The spent nuclear fuel would be shipped offsite at the end of Phase 1.

1B. Foreign research reactor spent nuclear fuel stored under analysis option 1A would be transferred to a newly constructed dry storage facility, where it would be stored until ultimate disposition. Foreign research reactor spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and stored at the new dry storage facility. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that would be considered for the Savannah River Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Savannah River Site would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 1A (above). Impacts of construction and operation of the dry storage facility considered in analysis option 1B would bound those of the facility required to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel at the Savannah River Site.
- Under Implementation Subalternative 1b (Section 2.2.2.1), the Savannah River Site would receive only HEU from the foreign research reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts

from the management of this amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A and 1B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Savannah River Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which in uranium content represents approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 1A or 1B (above) by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years, and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Savannah River Site would be bounded by analysis options 1A or 1B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and stored would remain constant. The impacts would be the same as in analysis options 1A or 1B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States because the foreign research reactors would consider their own alternatives as to whether or not to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel, in this case, cannot be quantified. The upper limit, however, as considered under analysis options 1A and 1B (above), would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the impacts at the Savannah River Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider wet storage technology for new construction. DOE would implement the policy by constructing a new wet storage facility at the H-Area or by using the BNFP, owned by Allied General Nuclear Services. DOE would have to acquire the facility which could be ready for use in approximately 5 years. Therefore, if the Savannah River Site is selected under either the Regionalization or Centralization Alternatives of the Programmatic SNF&INEL Final EIS, Phase 2 at the Savannah River Site could start as early as 5 years from the start of implementation period by using BNFP. The new wet storage facility is described in Section 2.6.5.1.2. BNFP is described in Section F.1. For this implementation alternative, an analysis option 1C is considered, which is similar to 1B, as follows:

1C. The spent nuclear fuel managed under analysis option 1A would be transferred to a newly constructed wet storage facility or the BNFP where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 would be received and managed at these facilities. For the purpose of this analysis, the amount of spent nuclear fuel that would be managed in these facilities would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in Section 2.3.6, the Savannah River Site is currently limited to chemical separation of aluminum-based foreign research reactor spent nuclear fuel.

Under Management Alternative 2, as discussed in Section 2.3, DOE and the Department of State would assess the management of foreign research reactor spent nuclear fuel in a foreign location which would include an evaluation of foreign reprocessing with acceptance by the United States of the vitrified high-level waste resulting from reprocessing. The waste would be received and managed at the Defense Waste Process Facility at the Savannah River Site. DOE estimates that the total volume of the vitrified high-level waste would be about 2.4 m^3 (8.5 ft^3) and it would fill about 16 European-size canisters. A European-size canister is about four times smaller than the canister used in the Defense Waste Process Facility at the Savannah River Site.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Savannah River Site would receive the aluminum-based fuel which would not be reprocessed overseas. This spent nuclear fuel would be processed at the Savannah River Site chemical separation facilities in the same manner as in Implementation Alternative 6 above. The amount of foreign research reactor aluminum-based spent nuclear fuel to be chemically separated would be approximately 12,200 elements, 12.9 MTHM, 79 m^3 (2,600 ft^3).

F.4.1.1 Existing Facilities (Phase 1)

Analysis option 1A utilizes existing facilities that would be ready to receive and store foreign research reactor spent nuclear fuel by late 1995. The environmental impacts from this analysis option include only those related to operations, specifically: socioeconomics; occupational and public health and safety; materials, utilities, and energy; air quality; and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at the management site is the maximum, and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.1.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 1A would be attributable to the staffing requirements for existing facilities. Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be small. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Savannah River Site. Use of the current work force would not result in any net socioeconomic impact relative to baseline employment data. In fact, using the current work force may partially compensate for the decline in employment expected from changes in site mission from 20,000 persons in 1995 to approximately 15,800 persons in 2004 (DOE, 1995g).

F.4.1.1.2 Occupational and Public Health and Safety

Radiological exposures could affect occupational and public health and safety. Possible sources of radiological exposure from the receipt and storage of foreign research reactor spent nuclear fuel include: (1) airborne emissions from incident-free operations; (2) incident-free handling activities; and (3) airborne emissions from accident conditions. Radiological exposures are presented in individual subsections for emissions-related impacts and handling-related impacts. Accident-related impacts are presented in Section F.4.1.3.

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to airborne emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the maximally exposed individual (MEI), defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the existing storage facility (RBOF and/or L-Reactor disassembly basin) during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-21 summarizes the annual emission-related doses to the public and the associated risks for the MEI and the population at the Savannah River Site during Phase 1 operations.

Table F-21 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Savannah River Site (Phase 1)

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• RBOF (wet)	0.00011	5.5×10^{-11}	0.0057	0.0000028
• L-Reactor Basin (wet)	0.000073	3.7×10^{-11}	0.0046	0.0000023
Storage at:				
• RBOF (wet)	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
• L-Reactor Basin (wet)[a]	0.00036	1.8×10^{-10}	0.022	0.000011

[a] *L-Reactor basin doses are due to existing conditions. The foreign research reactor spent nuclear fuel contribution would be six orders of magnitude lower.*

Handling-Related Impacts: Management site workers would receive radiation doses during handling operations, such as receiving and unloading the transportation casks, transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1A involves the receipt of 644 shipments of spent nuclear fuel into the existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, and the preparation of 161 transportation casks for offsite shipment at the end of Phase 1. It was assumed that at the end of a 10-year period (i.e., Phase 1), the spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodology used to calculate the doses to a working crew associated with the handling activities of the spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculation. However, the upper bound for such a dose would be equal to administrative or regulatory limits at the management site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If one worker received the full 5,000 mrem per year for the full 13 years of potential spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. The associated risk of incurring a latent cancer fatality (LCF) would be 2.6 percent. The collective dose to the workers would be 250 person-rem with an associated LCF risk of 0.10.

F.4.1.1.3 Material, Utility, and Energy Requirements

The estimated annual consumption of materials, utilities, and energy from the use of existing storage facilities is shown in Table F-22.

Table F-22 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Commodity	*Baseline Site Usage*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	1,430	784	0 percent
Fuel (l/yr)	28,400,000	6,570	15,000	0.05 percent
Water (l/yr)	88,200,000,000	35,100,000	2,900,000	0.04 percent

The material, utility, and energy requirements for analysis option 1A would represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in fuel consumption at the Savannah River Site would be minimal because overall onsite activity would not increase due to changes in the Savannah River Site mission and the general reduction in employment levels. The Programmatic SNF&INEL Final EIS concluded that the existing capacities and distribution systems for electricity, steam, water, and domestic wastewater treatment are adequate to support any of the five alternatives considered for spent nuclear fuel management at the Savannah River Site. This conclusion would also be valid for analysis option 1A because it is bounded by the alternatives considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

F.4.1.1.4 Waste Management

The estimated annual waste generation for foreign research reactor spent nuclear fuel at the RBOF and L-Reactor disassembly basin is shown in Table F-23. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by foreign research reactor spent nuclear fuel storage at the RBOF and L-Reactor disassembly basin. Therefore, the impact of this waste on existing Savannah River Site waste management activities would be minimal.

Table F-23 Annual Waste Generation for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Waste Form	*Baseline Site Generation*	*RBOF*	*L-Reactor Basin*	*Percent Increase*
High-Level Waste (m^3/yr)	127,400[a]	0	0	0 percent
Transuranic Waste (m^3/yr)	760	0	0	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	161	510	2.6 percent
Wastewater (l/yr)	690,000,000	2,650,000	35,000,000	5.1 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

F.4.1.1.5 Air Quality

Nonradiological Emissions: Impact assessments for nonradiological air emissions associated with implementation of the respective spent nuclear fuel management alternatives (excluding construction-related activities) are based primarily on analyses performed for the Programmatic SNF&INEL Final EIS (DOE, 1995g). These analyses were based on the following assumptions and qualifications:

- Air emissions data for wet storage are based upon releases from the RBOF
- All air pollutant sources, except standby diesel generators, are assumed to operate continuously. The standby diesel generators are assumed to operate daily for 1 hour; whereas their actual operation consists of a single monthly test.

Table F-24 lists the annual potential maximum emissions of criteria and toxic air pollutants (in tons per year) attributable to existing facilities. These data indicate little or no difference in pollutant loading between the baseline and the regionalization alternative considered in the Programmatic SNF&INEL Final EIS. The greatest pollutant contribution for criteria air pollutants would be for nitrogen oxides (7.7 metric tons per year, or 8.5 tons per year) and carbon monoxide (1.8 metric tons per year, or 2.0 tons per year). Assuming that the foreign research reactor spent nuclear fuel comprises approximately nine percent of the total spent nuclear fuel managed under the Centralization Alternative of the Programmatic SNF&INEL Final EIS, the incremental and cumulative nonradiological air quality impacts attributable to the storage of foreign research reactor spent nuclear fuel in existing facilities (RBOF, L-Reactor disassembly basin) would be small.

Table F-24 Annual Maximum Emissions of Criteria Air Pollutants Attributable to Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Savannah River Site (Phase 1)

Pollutant	*Baseline Wet (tons/yr)*[a,b,c]	*RBOF (tons/yr)*[a,b]	*Percent Increase*
Sulfur Oxides	0.4	0.005	1.3 percent
Nitrogen Oxides	6.0	0.77	12.8 percent
Total Suspended Particulates	0.4	0.006	1.5 percent
Carbon Monoxide	1.5	0.18	12 percent
Total Volatile Organic Compounds	0.6	0.077	12.8 percent
Gaseous Fluorides	none	none	none

[a] *Source: Hunter and Stewart, 1994*

[b] *To convert tons to metric tons, multiply by 0.907*

[c] *Decentralization based on management of the existing Savannah River Site inventory of spent nuclear fuel*

Radiological Emissions: The ventilation system serving the RBOF is designed to minimize airborne radioactivity levels both inside and outside of the facility. This ventilation system is based on a "once through" multiple-air-zone concept in which air flows from areas with low potential for contamination to areas with higher potential. The airflow is passed through High-Efficiency Particulate Air filters mounted on the building exhaust. A review of 1993 emissions data from the RBOF indicates emissions of approximately 2.7×10^{-7} Ci per year of ^{137}Cs (DOE, 1995g).

The RBOF and L-Reactor disassembly basin are currently being utilized to wet store spent nuclear fuel, and their emissions are reflected within baseline environmental conditions.

F.4.1.1.6 Water Resources

The use of RBOF and the L-Reactor disassembly basin for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water usage at these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Viable accidents during this interim storage period could be a release of pool water onto the ground surface or a breach of the liner of the wet storage basins in which the spent nuclear fuel would be stored. These type of accidents have been analyzed for both the RBOF and the L-Reactor disassembly basin in the safety analysis documentation (Dupont, 1983a and 1983b; WSRC, 1995b and 1995c) and the Programmatic SNF&INEL Final EIS (DOE, 1995g). As discussed in the Programmatic SNF&INEL Final EIS, radionuclides in the released water would enter the water table aquifer but would not reach any surface-water or any drinking water aquifer on or off the Savannah River Site. Basin water contains no toxic or hazardous chemicals, therefore, accidental releases from the basins would have minimal impacts on surface- and groundwater resources.

F.4.1.2 New Facilities (Phase 2)

Analysis options 1B and 1C involve the use of new facilities for the storage of foreign research reactor spent nuclear fuel at the Savannah River Site. The environmental impacts analyzed relate to the construction and operation of these new facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities and energy; and waste management.

F.4.1.2.1 Dry Storage

Analysis option 1B is associated with the construction and operation of new dry storage facilities. The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 of this EIS and earlier in this appendix. None of the environmental impact parameters discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.1.2.1.1 Land Use

A new dry storage facility would be located in one of two 60-plus ha (150-plus acres) undeveloped areas near the H- and P-areas, respectively. Predominant land use at both areas is managed timber land. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. This represents about 6 percent of the available space at either area. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at either area would significantly impact land use patterns on the Savannah River Site.

F.4.1.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents about 7.7 percent of the estimated FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs from a new dry storage facility are estimated to be $15.6 million for receipt and handling and $.6 million for storage. These costs represent about 1.3 percent and 0.05 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Savannah River Site.

F.4.1.2.1.3 Cultural Resources

There are no known cultural or historic resources located within the two proposed construction locations for a new dry storage facility. Both locations are within an area of low archaeological site density. Activities within this zone would have a low probability of encountering archaeological sites and virtually no chance of impacting large sites with more than three prehistoric components. Neither location has been specifically surveyed for archaeological resources, but this would occur prior to initiation of any construction-related activities.

Three Native American groups have expressed concerns relating to the possible existence on the Savannah River Site of several plant species traditionally used in Tribal ceremonies. These plant species are known to occur on the Savannah River Site, typically in wet, sandy areas such as evergreen shrub bogs and savannas. However, these plants are not likely to be found in the two proposed construction locations because of a lack of suitable habitat.

F.4.1.2.1.4 Aesthetics and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the new dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.1.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Savannah River Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres).

Site preparation, land shaping and grading activities associated with construction would present a slight to moderate erosion hazard, which would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the Savannah River Site.

F.4.1.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction-related activities include the generation of fugitive dust (particulate matter), smoke from earth moving and clearing operations, and emissions from the construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Cleared vegetation would be burned at the construction site rather than hauled to a landfill. The open burning of this material is not expected to adversely impact ambient air quality at the Savannah River Site. As shown in Table F-25, air quality impacts associated with construction-related activities would be minimal and compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-25 Estimated Maximum Concentrations of Criteria Pollutants at the Savannah River Site Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*[b]	*Construction Activities*
Savannah River Site Boundary ($\mu g/m^3$):				
• Total Suspended Particulate (TSP)	Annual	75	11	0.002 - 0.003
• Particulate Matter (PM_{10}), Daily	24-hr	150	56	0.1
• Particulate Matter (PM_{10}), Daily	Annual	50	2.7	0.003

[a] *Source: (DOE, 1995g)*

[b] *Baseline values due to actual emissions from all the Savannah River Site sources during 1990 plus sources permitted through 1992*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions associated with new dry storage would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a dry new storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively high contamination) would help provide contamination control. High-Efficiency Particulate Air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depend on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a vapor state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

Radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions discussed in Section F.6. The radiological consequences of air emissions are discussed in Section F.4.1.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.1.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent no more than a 0.002 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.1.2.1.8 Ecology

Terrestrial Resources: The two proposed locations for new spent nuclear fuel management facilities encompass approximately 60-plus ha (150-plus acres) of undeveloped forest land. Surface vegetation consists primarily of upland pine stands. Loblolly and slash pine dominate, but small pockets of hardwoods (oaks, hickory, sweetgum, and yellow poplar) are also evident. The locations possess suitable habitat for white-tailed deer and feral hogs, as well as other faunal species common to the mixed pine/hardwood forests of South Carolina. The locations contain no suitable habitat for the various threatened and endangered species found on the Savannah River Site. The construction of a new dry storage facility would necessitate the clearing of 3.7 ha (9 acres) and is therefore not expected to significantly affect the terrestrial ecology of the area.

Wetlands: Dry storage of foreign research reactor spent nuclear fuel would not adversely impact wetlands. Although two small wetland areas are located along the southeastern perimeters of the potential storage locations, there is sufficient land area available within these locations to avoid these critical habitats. The implementation of best-management practices to control surface runoff and sedimentation would ensure the protection of wetlands and the aquatic ecosystem during construction activities.

Threatened and Endangered Species: The potential locations contain no suitable habitat for threatened, endangered, or candidate species known to occur on or near the Savannah River Site (DOE, 1995g). The southern bald eagle and wood stork feed and nest near wetlands, streams, and reservoirs, and thus would not be attracted to the highly industrialized foreign research reactor spent nuclear fuel management sites. Red-cockaded woodpeckers prefer open pine forests with mature trees greater than 70 years old for nesting and 30 years old for foraging. It is not believed that this species utilizes the relatively young pine stands (5 to 40 years of age) present within the potential storage locations. The nearest red-cockaded woodpecker colony is located across Upper Three Runs Creek, approximately 3.2 km (2 mi) north of H-Area. DOE has begun consultations with the U.S. Fish and Wildlife Service to determine the potential for endangered species to be affected, as required by the Endangered Species Act. Impacts to threatened and endangered species are not anticipated.

F.4.1.2.1.9 Noise

Noise generated onsite by construction or operation of a new dry storage facility should not adversely affect the public or the Savannah River Site environment. Noise generated by construction would be site specific and short lived. A small number of new construction jobs would be generated, but the resultant temporary increase in worker and materials traffic is expected to be insignificant compared to existing baseline traffic loads. Noise generated by operation would not significantly impact the environment because the facility would be located adjacent to previously developed, industrialized areas. The number of daily freight trains in the region and through the site (approximately 13) are not expected to change as a result of dry storage. There may be a slight increase in truck traffic to and from the potential storage locations, but this is not expected to result in a perceptible increase in traffic noise or any change in community reaction to noise along the major access routes to the Savannah River Site.

F.4.1.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Savannah River Site. Again, traffic congestion would not be a significant problem. As long as commercial trucks are complying with the Federal and State loading and speed regulations, truck traffic would not significantly damage the roadbed.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing the Savannah River Site labor force.

F.4.1.2.1.11 Occupational and Public Health and Safety

Emissions Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-26 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-26 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Savannah River Site (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Dry Storage Facility	0.00018	9.0×10^{-11}	0.0086	0.0000043
Storage at: • New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports to the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Doses were calculated for the dry vault and dry cask designs. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-27 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-27 Handling-Related Impacts to Workers at the Savannah River Site (New Dry Storage)

	Worker Population Dose (person-rem)			*Worker Population Risk (LCF)*		
	RBOF/L-Reactor	*New Dry Storage Cask*	*New Dry Storage Vault*	*RBOF/L-Reactor*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phase 1	250	NA	NA	0.10	NA	NA
Phases 1 and 2	NA	416	363	NA	0.17	0.15

NA = Not Applicable

F.4.1.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Savannah River Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel; and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-28. These requirements represent a small percent of current requirements for the Savannah River Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Savannah River Site is expected to decrease due to changes in site mission and a general reduction in employment.

Table F-28 Annual Utility and Energy Requirements for New Dry Storage at the Savannah River Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	800 - 1,000	0.15 percent
Fuel (l/yr)	28,400,000	0	0 percent
Water (l/yr)	88,200,000,000	1,590,000[a] 400,000[b]	0.002 percent[a] 0.00046 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.1.2.1.13 Waste Management

Construction of a new dry storage facility at the Savannah River Site would generate approximately 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-29. These quantities represent a very small percent increase above current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

Table F-29 Annual Waste Generated from New Dry Storage at the Savannah River Site

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level Waste (m^3/yr)	127,400[a]	none	0 percent
Transuranic Waste (m^3/yr)	760	none	0 percent
Solid Low-Level Waste (m^3/yr)	19,750	22[b] 1[c]	0.11 percent[b] 0.005 percent[c]
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.21 percent[b] 0.06 percent[c]

[a] *Total inventory (m^3) at the Savannah River Site*

[b] *During receipt and handling*

[c] *During storage*

F.4.1.2.2 Wet Storage

Analysis option 1C is associated with the construction and operation of a new wet storage facility or the modification and operation of BNFP at the Savannah River Site (Implementation Alternative 5 to Management Alternative 1). The environmental impacts from the modification of the BNFP would be bounded by the impacts associated with the construction of a new wet storage facility.

F.4.1.2.2.1 Land Use

A new wet storage facility would be located in one of two 60-plus ha (150-plus acres) undeveloped areas near the H- and P-areas, respectively. Predominant land use at both areas is managed timber land. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents less than 5 percent of the available space at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Savannah River Site.

F.4.1.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be \$449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$112.2 million. This represents approximately 9.4 percent of the estimated FY 1995 total expenditures for the Savannah River Site (1,198 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be small but positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 1.9 percent and 0.3 percent of FY 1995 total expenditures for the Savannah River Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Savannah River Site of approximately 20,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Savannah River Site.

F.4.1.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.1.2.1.3).

F.4.1.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.1.2.1.4).

F.1.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.1.2.1.5).

F.4.1.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 2.8 ha (7 acres) of land. In comparison, approximately 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with new dry storage construction, as presented in Table F-25.

Operations-related impacts associated with wet storage would be similar to those discussed under existing facilities.

Radiological Emissions: Incident-free airborne releases from a new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3-micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. The truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.1.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 88,200 million l (23,300 million gal) for the Savannah River Site, these amounts represent an increase of less than 0.01 percent for both. Therefore, a new wet storage facility would have minimal impact on water resources at the Savannah River Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Savannah River Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Savannah River Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Savannah River Site would still be well within the design capacities of treatment systems at the Savannah River Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.1.2.2.8 Ecology

Impacts to the ecology would be the same as for new dry storage (Section F.4.1.2.1.8).

F.4.1.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.1.2.1.9).

F.4.1.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.1.2.1.10).

F.4.1.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Savannah River Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-30 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Savannah River Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-30 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• BNFP	0.00065	3.3×10^{-10}	0.0045	0.0000023
• New Wet Storage Facility	0.00011	5.5×10^{-11}	0.0057	0.0000028
Storage at:				
• BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
• New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 1C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into an existing wet storage facility (RBOF and/or L-Reactor disassembly basin) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-31 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Savannah River Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-31 Handling-Related Impacts to Workers at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*Worker Population Dose (person-rem)*	*Worker Population Risk (LCF)*
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phase 1 and Phase 2: New Wet Storage Facility	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phase 1 and Phase 2: BNFP	360	0.14
Phase 1: RBOF/L-Reactor Basin	250	0.10
Phase 1 and Phase 2: BNFP[a]	310	0.12

[a] *Assumes that BNFP would be ready in 5 years instead of 10 years.*

F.4.1.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Savannah River Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-32. These requirements represent a small percent of current requirements for the Savannah River Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Savannah River Site is expected to decrease due to changes in site mission and a general reduction in employment.

Table F-32 Annual Utility and Energy Requirements for New Wet Storage at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	659,000	1,000 - 1,500	0.23 percent
Fuel (l/yr)	28,400,000	0	0 percent
Water (l/yr)	88,200,000,000	2,700,000[a] 1,500,000[b]	0.003 percent 0.001 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.1.2.2.13 Waste Management

Construction of a new wet storage facility would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-33. These quantities represent a very small percent increase in current levels at the Savannah River Site. Existing waste management storage and disposal activities at the Savannah River Site could accommodate the waste generated by a new wet

Table F-33 Annual Waste Generated from New Wet Storage at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level (m^3/yr)	127,400[a]	none	0 percent
Transuranic (m^3/yr)	760	none	0 percent
Solid Low-Level (m^3/yr)	19,750	16[b] 1[c]	0.08 percent 0.005 percent
Wastewater (l/yr)	690,000,000	1,590,000[b] 400,000[c]	0.23 percent 0.06 percent

[a] *Total inventory (m^3) at the Savannah River Site.*

[b] *During receipt and handling*

[c] *During storage*

storage facility. Therefore, the impact of this waste on the existing Savannah River Site waste management capacities would be minimal.

F.4.1.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Savannah River Site is presented here, based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated.

The radiation doses to the following individuals and the general population are calculated for accident conditions at the spent nuclear fuel storage facility:

- ***Worker*:** An individual located 100 m (330 ft) from the radioactive material release point. For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (330 ft) for elevated release. The direction to the worker was chosen as the direction to the maximally exposed sector. The dose to the worker is calculated for the 50th-percentile meteorological condition (DOE, 1992a).

- ***Maximally Exposed Offsite Individual (MEI):*** A theoretical individual living at the storage site boundary receiving the maximum exposure. The individual is assumed to be located in a direction downwind from the release point. The dose to the MEI is shown for the 95th-percentile meteorological condition.

- ***Nearest Public Access Individual (NPAI).*** An individual stranded on a highway or public access road near to the facility at the time of an accident. The distance to the NPAI was chosen as the distance to the nearest public access point; the direction was chosen as the direction to that point. The dose to the NPAI is shown for the 95th-percentile meteorological condition.

- ***General Population Within an 80 km (50 mi) Radius of the Facility:*** The dose calculations are performed for the direction downwind from the release point that results in highest dose to the public. The dose to the population is shown for the 95th-percentile meteorological condition.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using external (direct exposure), inhalation, and ingestion pathways. Dispersion in air from point of release was estimated with both 50th-percentile and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition. The 95th-percentile condition is defined as that condition which is not exceeded more than 5 percent of the time, and is more conservative than the 50th-percentile condition.

The ingestion dose is calculated by considering that the individual and the public would consume contaminated food produced in the vicinity [up to 80 km (50 mi)] of the accident. This is conservative, and it is expected that continued consumption of contaminated food products by the public would be suspended after a protective action guideline is reached. In 1991, the U.S. Environmental Protection Agency recommended protective action guidelines in the range of one to five rem whole-body exposure (EPA, 1991). To ensure a consistent analytical basis, no reduction of exposure due to a protective action guideline was used in this analysis.

Accidents considered for detailed analysis are similar to those analyzed in the Programmatic SNF&INEL Final EIS. The selection of accidents was based on the following considerations:

- Accidents in the Programmatic SNF&INEL Final EIS were reviewed to select reasonably foreseeable accidents. They are: (1) criticality caused by human error during operation, equipment failure, or earthquake; (2) mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a spent nuclear fuel element); and (3) accident involving an impact by either an internal or an external initiator with and without an ensuing fire.

Six accident scenarios were evaluated at each storage location using identical source terms (estimated amounts of radioactive material released during postulated accidents). The wet pool accidents are assumed to be cutting into the fuel region or mechanical damage due to operator error, an accidental criticality, and an aircraft crash into the water pool facility. The dry storage accidents are assumed to be cutting into the fuel region or mechanical damage during examination work and handling in a dry cell, dropping of a fuel cask, and an aircraft crash with an ensuing fire.

Table F-34 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of the policy at each site to obtain conservative estimates of risks for the entire program at the Savannah River Site. These risk estimates are presented in Table F-35.

Table F-36 presents the frequency and consequences of the accidents analyzed for the Savannah River Site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at the Savannah River Site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Savannah River Site. These annual risks are multiplied by the maximum duration of this implementation alternative at the Savannah

Table F-34 Frequency and Consequences of Accidents at the Savannah River Site

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents - New					
• Spent Nuclear Fuel Assembly Breach	0.16	0.24	0.068	9.2	28
• Dropped Fuel Cask	0.0001	0.018	0.00034	0.55	0.28
• Aircraft Crash w\Fire	1×10^{-6}	40	0.29	1300	120
Wet Storage Accidents at RBOF					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	0.00039	0.23	0.14
• Accidental Criticality	0.0031	130	44	4,800	16,000
• Aircraft Crash	1×10^{-6}	4.1	0.98	150	400
Wet Storage Accidents at L-Reactor Basin					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0093	0.00097	0.14	0.11
• Accidental Criticality	0.0031	170	120	3,000	14,000
• Aircraft Crash	1×10^{-6}	4.2	2.6	93	70

Table F-35 Annual Risks of Accidents at the Savannah River Site

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents - New				
• Spent Nuclear Fuel Assembly Breach	1.9×10^{-8}	5.5×10^{-9}	0.00075	0.0000018
• Dropped Fuel Cask	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	1.1×10^{-11}
• Aircraft Crash w\Fire	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.8×10^{-11}
Wet Storage Accidents at RBOF				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.0×10^{-7}	7.0×10^{-8}	0.0074	0.000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
Wet Storage Accidents at L-Reactor Basin				
• Spent Nuclear Fuel Assembly Breach	7.4×10^{-10}	8.0×10^{-11}	0.000011	7.1×10^{-9}
• Accidental Criticality	2.6×10^{-7}	1.9×10^{-7}	0.0047	0.000017
• Aircraft Crash	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	2.8×10^{-11}

Table F-36 Frequency and Consequences of Accidents at Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Wet Storage Facility - New					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0070	0.00039	0.23	0.14
• Accidental Criticality	0.0031	17	9.5	370	1,600
• Aircraft Crash	1×10^{-6}	4.1	0.98	150	400
BNFP					
• Spent Nuclear Fuel Assembly Breach[a]	0.16	0.018	0.00099	0.028	0.0008
• Accidental Criticality[a]	0.0031	80	75	44	75
• Aircraft Crash	1×10^{-6}	92	31	23	70

[a] *Emissions would be released through a tall stack.*

River Site to obtain conservative estimates of risks at the Savannah River Site. Table F-37 presents the risk estimates from this implementation alternative.

Table F-37 Annual Risks of Accidents at the Savannah River Site (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Wet Storage Facility - New				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-10}	3.1×10^{-11}	0.000019	8.8×10^{-10}
• Accidental Criticality	2.7×10^{-8}	1.5×10^{-8}	0.00060	0.0000020
• Aircraft Crash	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	1.6×10^{-10}
BNFP				
• Spent Nuclear Fuel Assembly Breach[a]	2.8×10^{-9}	8.0×10^{-11}	0.0000023	5.2×10^{-11}
• Accidental Criticality[a]	1.3×10^{-7}	1.2×10^{-7}	0.000070	9.2×10^{-8}
• Aircraft Crash[a]	4.6×10^{-10}	1.6×10^{-11}	1.2×10^{-8}	2.8×10^{-10}

[a] *Emissions would be released through a tall stack.*

F.4.1.3.1 Secondary Impact of Radiological Accidents at the Savannah River Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the results of the accident calculations presented in Section F.4.1.3. Radiological accidents that would result in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Savannah River Site, the radiological accident with the highest MEI dose is the accidental criticality at a wet storage facility (Table F-34). For this accident, the MEI dose would be 170 mrem. For the air immersion and ground surface pathways only, the dose would be 50 mrem, (Table F-115A) which is lower than the 100 mrem limit used in this analysis. Local contamination would be likely around the dry storage facility, but is expected to be contained entirely within the boundaries of the Savannah River Site. Cleanup activities should be small and any impacts to land uses, cultural resources, water quality, and ecology would be reversible. No impacts to national defense or local economies would be expected.

F.4.1.4 Cumulative Impacts at the Savannah River Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and other offsite (non-DOE) facility impacts at the Savannah River Site. A major portion of the presentation is based on information included in the Interim Management of Nuclear Materials Final EIS for the Savannah River Site, issued in October 1995 (DOE, 1995b). The cumulative impacts include those associated with the handling and dry storage of foreign research reactor spent nuclear fuel at the Savannah River Site and the following existing or major foreseeable activities proposed for the site:

- The operation of the Vogtle Electric Generating Plant located approximately 16 km (10 mi) south west of the center of the Savannah River Site.
- The implementation of the preferred scenario in the Interim Management of Nuclear Materials EIS (DOE, 1995b).
- Shipment of aluminum-based spent nuclear fuel to the Savannah River Site for storage and disposal discussed in Appendix C of the Programmatic SNF&INEL Final EIS (DOE, 1995g).
- Completion of the construction and operation of the Defense Waste Processing Facility (DOE, 1994g).
- Processing of F-Canyon plutonium solutions to metal (DOE, 1994a).
- Treatment and minimization of radioactive and hazardous wastes at the site as identified in the Savannah River Site Waste Management Final EIS (DOE, 1995f).
- Construction of an accelerator for tritium production at the Savannah River Site, along with associated support facilities (DOE, 1995a).
- Disposition of Surplus Highly Enriched Uranium at the site (DOE, 1995e).
- Storage and Disposition of Weapons-Usable Fissile Materials.
- Stockpile Stewardship and Management Program.
- Current Savannah River Site projects (based on 1993 data).

Any other foreseeable activities would have minimal impacts compared to the activities considered above.

Table F-38 summarizes the cumulative impacts for land use, socioeconomics, nonradiological air quality, occupational and public health and safety, waste generation, and energy and water consumption. As shown in Table F-38, the contribution of foreign research reactor spent nuclear fuel to the cumulative impacts at the Savannah River Site would be minimal.

F.4.1.5 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities for the receipt and management of foreign research reactor spent nuclear fuel at the Savannah River Site would result in some adverse impacts to the environment. Changes in designs and other methods of mitigation could eliminate, avoid, or reduce most impacts to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but could be controlled by water and dust suppressants. Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

The maximum loss of habitat would result from conversion of approximately 4 ha (10 acres) of managed pine forest to industrial land use.

Table F-38 Cumulative Impacts at the Savannah River Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Current Activities*[a]	*Other Activities*[b]	*Cumulative Impact*
Land Use (acres)	9	9,075[c]	3,975	13,059
Socioeconomics (persons)	190[d]/30[e]	(f)	11,000[d]/6,200[e]	11,190[d]/6,230[e]
Air Quality (nonradiological)	See Table F-38A	See Table F-38A	See Table F-38A	See Table F-38A
Occupational and Public Health and Safety:				
• MEI Dose (rem/yr)	3.6×10^{-7}	0.00025	0.0041	0.0043
LCF (per year)	1.8×10^{-10}	1.25×10^{-7}	0.000002	0.000002
• Population Dose (person-rem/yr)	0.022	9.1	295	304
LCF (per year)	0.000011	0.0045	0.15	0.154
• Worker Collective Dose (person-rem/yr)	10[g]	263	1,418	1,691
LCF (per year)	0.004	0.10	0.57	0.67
Energy and Water Consumption[j]				
• Electricity (MW-hr/yr)	1,000	659,000	4,104,106[h]	4,764,106
• Fuel (million l/yr)	0	28.4	3.06	31.47
• Steam (million kg/yr)	0	1,700	1,550	3,250
• Coal (tons/yr)	0	210,000	20,440	230,440
• Water (million l/yr)	2.2	88,200	6,796	94,996
Waste Generation				
• High-Level (m^3/yr)	0	(i)	6,330	6,330
• Low-Level (m^3/yr)	22	(i)	35,600	35,622
• Saltstone (m^3/yr)	0	(i)	60,000	60,000
• Transuranic (m^3/yr)	0	(i)	1,038	1,038
• Mixed/Hazardous (m^3/yr)	0	(i)	2,561	2,561

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Based on 1993 site data*

[b] *Other activities include: interim management of nuclear materials, spent nuclear fuel management, Vogtle plant operation, defense waste processing facility operations, stabilization of Pu-solutions, site-wide waste management activities, tritium accelerator facility, disposition of surplus HEU, storage and disposition of weapons-usable fissile materials, and the stockpile stewardship and management program activities*

[c] *Five percent of the total SRS site area of 181,500 acres*

[d] *Increase over baseline during construction activities*

[e] *Increase over baseline during operation activities*

[f] *Baseline working force approximately 20,600 persons*

[g] *The dose is due to the handling of the FRR SNF during receipt and transfer between facilities averaged over 40 years*

[h] *Major portion is the requirement for electricity by the tritium production accelerator facility (3,740,000 MW-hr/yr)*

[i] *Included in "other activities"*

[j] *During operation activities*

The amount of radioactivity that incident-free operation of the spent nuclear fuel facilities would release is a small fraction of the cumulative operational releases at the Savannah River Site and would be well below applicable regulatory standards (see Tables F-38 and F-38A).

Table F-38A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Savannah River Site Boundary[a]

Pollutant	Averaging Time	Regulatory Standard, (μg/m³)	Cumulative Concentration (μg/m³)[b]
Carbon Monoxide	1-hour	40,000	331.7 (0.83%)
	8-hour	10,000	52.3 (0.52%)
Nitrogen Oxides	Annual	100	19.5 (19.5%)
Sulfur Dioxide	3-hour	1,300	1,159 (89.1%)
	24-hour	365	248 (68.2%)
	Annual	80	17 (21.3%)
Gaseous Fluorides	12-hour	3.7	1.38 (37.3%)
	24-hour	2.9	0.58 (20%)
	1 week	1.6	0.56 (34.8%)
	1 month	0.8	0.066 (8.2%)
Nitric Acid	24-hour	125	9.8 (7.8%)

[a] *Concentrations represent: foreign research reactor spent nuclear fuel management, other DOE-owned spent nuclear fuel management, defense waste processing facility operations, consolidated incineration facility operation, stabilization of Pu-solutions, waste management activities, tritium supply and recycling, disposition of surplus highly enriched uranium, storage and disposition of weapons-usable fissile materials, and stockpile and stewardship management program activities*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

F.4.1.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Savannah River Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery. Construction and operation of facilities for foreign research reactor spent nuclear fuel management would also require the withdrawal of water from surface- and groundwater sources, but most of this water would return to onsite streams or the Savannah River after use and treatment.

F.4.1.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Savannah River Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Savannah River Site would be minimal in most environmental media.

Pollution Prevention: DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements;" Executive Order 12780, "Federal Acquisition, Recycling and Waste Prevention;" and applicable DOE orders and guidance documents in planning and implementing pollution prevention at the Savannah River Site. The pollution prevention program at the Savannah River Site was initiated in 1990 as a waste minimization program. Currently, the program consists of four major initiatives: solid waste minimization, source reduction and recycling of wastewater discharges, source reduction of air emissions, and potential procurement of products manufactured from recycled materials. Since 1991, waste (all types) generated at the Savannah River Site has decreased, with the greatest reductions in hazardous and mixed wastes. These reductions are attributable primarily to material substitutions (DOE, 1995g).

All foreign research reactor spent nuclear fuel activities at the Savannah River Site would be subject to a pollution prevention program. Implementation of the program plan would minimize waste generated by these activities (DOE, 1995g).

Cultural Resources: A Programmatic Memorandum of Understanding, ratified on August 24, 1990, between the DOE Savannah River Operations Office, the South Carolina State Historic Preservation Office, and the Advisory Council on Historic Preservation is the instrument for the management of cultural resources at the Savannah River Site. DOE uses this memorandum to identify cultural resources and develop mitigation plans for affected resources in consultation with the State Historic Preservation Office.

DOE would comply with the terms of the memorandum in support of foreign research reactor spent nuclear fuel activities at the Savannah River Site. For example, DOE would survey sites prior to disturbance and reduce impacts to any potentially significant resources discovered through avoidance or removal. Any artifacts encountered would be protected from further disturbance and the elements until removed (DOE, 1995g).

DOE conducted an investigation of Native American concerns over religious rights in the Central Savannah River Valley in conjunction with studies in 1991 related to a New Production Reactor. During this study, three Native American groups expressed concern over sites and items of religious significance on the Savannah River Site. DOE has included these organizations on its environmental mailing list, solicits their comments on NEPA actions on the Savannah River Site, and sends them documents about the Savannah River Site environmental activities, including those related to foreign research reactor spent nuclear fuel management considerations. These Native American groups would be consulted on any actions that may follow subsequent site-specific environmental reviews (DOE, 1995g).

Geology: DOE expects that there would be no impacts to geologic resources at the Savannah River Site under any storage option evaluated. Potential soil erosion in areas of ground disturbance would be minimized through sound engineering practices such as implementing controls for storm water runoff (e.g., sediment barriers), slope stability (e.g., rip-rap placement), and wind erosion (e.g., covering soil stockpiles). Relandscaping would minimize soil loss after construction was completed. These measures would be included in a site-specific Storm Water Pollution Prevention Plan that the Savannah River Site would prepare prior to initiating any construction (DOE, 1995g).

Air Resources: DOE would meet applicable standards and permit limits for all radiological and nonradiological releases to the atmosphere. In addition, the Savannah River Site would follow the DOE policy of maintaining radiological emissions to levels "as low as reasonably achievable" (ALARA). ALARA is an approach to radiation protection to control or manage exposures (both individual and collective) and releases of radioactive material to the environment as low as social, technical, economic, practical, and public policy considerations permit. ALARA is not a dose limit, but rather a process that has as its objective the attainment of dose levels as far below the applicable limits as practicable (DOE, 1995g).

Water Resources: DOE would minimize the potential for adverse impacts on surface water during construction through the implementation of a storm water pollution prevention plan that details controls for erosion and sedimentation. The plan would also establish measures for prevention of spills of fuel and chemicals and for rapid containment and cleanup (DOE, 1994g).

DOE could minimize water usage during both construction and operation of facilities by instituting water conservation measures such as instructing workers in water conservation (e.g., turn off hoses when not in use), installing flow restrictors, and using self-closing hose nozzles (DOE, 1995g).

Ecological Resources: DOE does not anticipate any impact on wetlands on the Savannah River Site as a result of the spent nuclear fuel program. In any case, it is DOE and the Savannah River Site policy to achieve "no net loss" of wetlands. Pursuant to this goal, DOE has issued a guidance document, "Information for Mitigation Impacts at the Savannah River Site," for project planners that puts forth a practical approach to wetlands protection that begins with avoidance of impacts (if possible), moves to minimization of impacts (if avoidance is impossible), and requires compensatory measures (wetlands restoration, creation, or acquisition) in the event that impacts cannot be avoided (DOE, 1995g).

The analysis indicates that there are no threatened or endangered species or sensitive habitats in the areas considered as representative of potential sites for foreign research reactor spent nuclear fuel activities at the Savannah River Site. However, DOE would perform site-specific predevelopment surveys to ensure that development of new facilities would not impact any of these biological resources (DOE, 1995g).

Noise: DOE anticipates that noise impacts both on and off the Savannah River Site would be minimal. DOE does not foresee noise impacts from the management of foreign research reactor spent nuclear fuel that would warrant mitigation measures beyond those consistent with good construction, engineering, operations, and management practices.

Traffic and Transportation: DOE has a system of onsite buses operating at the Savannah River Site. The Savannah River Site would evaluate the need for upgrades or changes in service that might be required for foreign research reactor spent nuclear fuel management activities and would make changes, as necessary.

DOE would manage changes in traffic volume or patterns during construction through such measures as designating routes for construction vehicles, providing workers with safety reminders, and upgrading onsite police traffic patrols, if necessary.

Occupational and Public Health and Safety: The DOE program for maintaining radiological emissions to levels "as low as reasonably achievable" would minimize any impacts to workers and the public due to atmospheric releases. Likewise, the Site Pollution Prevention Plan and emergency preparedness measures would enhance safety both on and off the Site (DOE, 1995g).

Accidents: The Savannah River Site has in place emergency action plans that would be activated in the case of an accident. These plans contain both onsite provisions (e.g., evacuation plans, response teams, medical and fire response, training and drills, communications equipment) and offsite arrangements (e.g., response plans for medical and fire agencies, coordination with local and State agencies, communication plans). The Savannah River Site plans would be updated to include any new facilities or activities related to spent nuclear fuel management that would involve the Savannah River Site. The execution of the plans in response to an accident would mitigate adverse effects both on the Savannah River Site and in all the surrounding areas (DOE, 1995g).

F.4.2 Idaho National Engineering Laboratory

If the Idaho National Engineering Laboratory is the site to manage all DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel would be received and managed at the site until ultimate disposition. If the Idaho National Engineering Laboratory is not the site to manage DOE-owned spent nuclear fuel, foreign research reactor spent nuclear fuel could be received and managed at the Idaho National Engineering Laboratory until the selected site(s) would be ready to receive the foreign research reactor spent nuclear fuel. The construction of new facilities for managing foreign research reactor spent nuclear fuel is estimated to take about 10 years; this period is referred to as Phase 1. The period following Phase 1 until ultimate disposition is referred to as Phase 2 (approximately 30 years). The amount of spent nuclear fuel that could be received at the Idaho National Engineering Laboratory under the basic implementation of Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, the Idaho National Engineering Laboratory could receive one-half of the foreign research reactor spent nuclear fuel under the Decentralization and the 1992/1993 Planning Basis alternatives, all of the TRIGA-type foreign research reactor spent nuclear fuel under the Regionalization by Fuel Type alternative, only the foreign research reactor spent nuclear fuel from Western ports under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As discussed in Section 2.6.4.1, the split of foreign research reactor spent nuclear fuel evenly between the Savannah River Site and the Idaho National Engineering Laboratory under the Decentralization and 1992/1993 Planning Basis alternatives in the Programmatic SNF&INEL Final EIS was not considered to have a practical basis, and was therefore not evaluated in detail.

As a potential Phase 1 site, the Idaho National Engineering Laboratory would receive and manage foreign research reactor spent nuclear fuel at existing dry and wet storage facilities. The existing facilities identified for this purpose would be the FAST facility in CPP-666, the IFSF in CPP-603, and the CPP-749 storage area. Descriptions of these facilities are provided in Appendix F, Section F.3.

The FAST facility is a modern underwater storage facility which has been used in the past for receipt and storage of foreign research reactor spent nuclear fuel. It has the capability to receive and unload spent nuclear fuel casks at a rate of approximately five per week. Storage capacity for up to 8,400 foreign research reactor spent nuclear fuel elements could be provided in a 10-year period by using the spent nuclear fuel storage racks that would be installed. The capability of the FAST facility to receive foreign research reactor spent nuclear fuel in the near term is limited due to the number of activities scheduled through FY 1998. Considering these activities, DOE estimates that 3,600 elements could be received by the end of 1999 at the FAST facility.

The IFSF is a shielded dry storage vault originally constructed for Fort St. Vrain reactor fuel. The storage capacity available is for approximately 9,000 foreign research reactor spent nuclear fuel elements. However, as with the FAST facility, many activities are already scheduled for the facility. Considering these activities, foreign research reactor spent nuclear fuel could not be received until sometime in FY 1997 and could continue at the rate of 50 shipments per year (approximately 1,500 elements) thereafter.

The CPP-749 underground spent nuclear fuel storage area is a dry storage facility with a remote unloading area and vault storage. With some refurbishment it could provide space for 3,600 elements starting in FY 1998 and 7,000 elements after FY 2002. The spent nuclear fuel would go through the IFSF to be placed in baskets and transferred to a compatible storage cask. The refurbishments of existing facilities are part of the ongoing programs at the site to be performed independent of the proposed action in this EIS.

Between these facilities there is sufficient storage space and handling capacity to accommodate the receipt and management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory during the Phase 1 period. The storage capacity available and estimated maximum receipt rate of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory was shown earlier in Figure F-18.

An additional option to enhance storage capacity during Phase 1 would be to use the existing facilities to unload the transportation casks, and provide storage capacity in dry storage casks which would be placed near the existing facility.

As a Phase 2 site, the Idaho National Engineering Laboratory would continue to receive and manage foreign research reactor spent nuclear fuel at existing facilities until a new dry storage facility becomes operational at the site. Foreign research reactor spent nuclear fuel managed at existing facilities would then be transferred to the new facility where it would remain until ultimate disposition. The new facility would also receive foreign research reactor spent nuclear fuel shipments directly from ports after Phase 1 concluded. Dry storage encompasses both the dry vault design and the dry cask design as described in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set which provides a typical, and in many cases, bounding estimate of the resulting impacts.

The specific analysis options under the basic implementation of Management Alternative 1 are as follows:

2A. The Idaho National Engineering Laboratory would receive foreign research reactor spent nuclear fuel during Phase 1 and manage it at the FAST, the IFSF, and/or the CPP-749 facilities. For the purpose of this analysis, the amount of fuel to be stored is all foreign research reactor spent nuclear fuel that would be received in a 10-year period (17,500 elements). The fuel would be shipped offsite at the end of Phase 1.

2B. Foreign research reactor spent nuclear fuel managed under analysis option 2A would be transferred to a newly constructed dry storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving at the United States after Phase 1 concludes would be received and managed at the new dry storage facility until ultimate disposition. For the purpose of this analysis, the amount of spent nuclear fuel that would be stored in the dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, as discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Idaho National Engineering Laboratory as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Idaho National Engineering Laboratory would be likely to receive and manage foreign research reactor spent nuclear fuel in existing facilities during the Phase 1 period. The impacts would be bounded by analysis option 2A above. The dry storage facility considered in analysis option 2B would be sized to accommodate this amount of fuel. The spent nuclear fuel would either be shipped offsite after Phase 1, or it would be managed along with the rest of the spent nuclear fuel that would be managed at the Idaho National Engineering Laboratory.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive only HEU from the reactors eligible under the policy. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A and 2B above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Idaho National Engineering Laboratory would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis options 2A or 2B by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. The impacts from the management of the decreased amount of spent nuclear fuel at the Idaho National Engineering Laboratory would be bounded by analysis options 2A or 2B above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis options 2A or 2B.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States because the foreign research reactors would consider their own alternatives about whether to send the spent nuclear fuel to the United States. The amount of foreign research reactor spent nuclear fuel in this case cannot be quantified. The upper limit, however, as considered under analysis options 2A or 2B, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the impacts at the Idaho National Engineering Laboratory.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Idaho National Engineering Laboratory for Phase 2 until ultimate disposition. The new wet storage facility is described in Section 2.6.5.1.2. For this implementation alternative, an analysis option 2C, which is similar to option 2B, is considered as follows:

2C. The spent nuclear fuel managed under option 2A would be transferred to a newly constructed wet storage facility where it would be managed until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would be received and managed at the new wet storage facility until ultimate disposition. For the purpose of this analysis, the amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. As noted in the discussion in Section 2.3.6, chemical separation of both aluminum-based and TRIGA foreign research reactor spent nuclear fuel is evaluated for the Idaho National Engineering Laboratory.

Under Management Alternative 3 (Hybrid Alternative), as discussed in Section 2.4, the Idaho National Engineering Laboratory would receive the foreign research reactor TRIGA spent nuclear fuel. This spent nuclear fuel would be managed at the Idaho National Engineering Laboratory in existing facilities until ultimate disposition. The amount of TRIGA spent nuclear fuel that would be stored is 4,900 elements, 1.0 MTHM, 19 m^3 (670 ft^3).

F.4.2.1 Existing Facilities (Phase 1)

Analysis option 2A utilizes existing facilities for receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. The impacts from this analysis option include only those related to operations, specifically: socioeconomics, occupational and public health and safety, utilities and energy, air quality, and waste management. For this analysis, it was assumed that the amount of foreign research reactor spent nuclear fuel to be received at this management site is the maximum and the receipt rate is uniform at approximately 1,800 elements per year.

F.4.2.1.1 Socioeconomics

Potential socioeconomic impacts associated with analysis option 2A would be attributable to staffing requirements at existing facilities (FAST and IFSF). Currently, these facilities are being used to store spent nuclear fuel, so any incremental staffing requirements related to foreign research reactor spent nuclear fuel storage would be insignificant. All personnel required for the operation and support of the existing facilities could be acquired from the current work force at the Idaho National Engineering Laboratory. Use of the current work force would not result in any net socioeconomic impact relative to baseline environmental conditions. In fact, using the current work force would partially compensate for the decline in employment expected from changes in site mission.

F.4.2.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage would take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-39 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-39 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage in Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• IFSF/CPP-749 (dry storage)	0.00056	2.8×10^{-10}	0.0045	0.0000023
• FAST (wet storage)	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at:				
• IFSF/CPP-749 (dry storage)	0	0	0	0
• FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2A involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into existing storage facilities (IFSF/CPP-749 and FAST) during Phase 1, and the preparation of 161 transportation casks for shipment at the end of Phase 1. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of the analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Calculations were performed for both dry and wet existing storage facilities. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

The collective doses that would be received by the members of the working crew and the associated risk were calculated for Phase 1 operations. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent. The collective dose to the workers handling the transportation casks is 257 person-rem at the dry storage facilities and 250 person-rem at the wet storage facilities. The associated risk of incurring an LCF is 0.10.

F.4.2.1.3 Material, Utility, and Energy Requirements

The material, utility, and energy requirements at the FAST and IFSF are typical of those for wet storage and dry storage, respectively. They are discussed in more detail in Sections F.4.2.2.1.12 and F.4.2.2.2.12. Table F-40 summarizes the estimated annual requirements for these technologies.

Table F-40 Annual Utility and Energy Requirements for Foreign Research Reactor Spent Nuclear Fuel Storage at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Commodity	Baseline Site Usage	FAST	IFSF	Percent Increase
Electricity (MW-hr/year)	208,000	1,490	1,490	0.72 percent
Water (l/year)	6,500,000,000	1.93 million	2.12 million	0.033 percent
Fuel (l/year)	11,123,400	0	0	0 percent

The requirements for all storage options represent a small percentage of current requirements. No new generation or treatment facilities would be necessary. Increases in the Idaho National Engineering Laboratory fuel consumption would be minimal because overall activity would not increase due to changes in the Idaho National Engineering Laboratory mission and the general reduction in employment levels. The overall impacts of any of the storage options at the Idaho National Engineering Laboratory on materials, utilities, and energy resources would be minimal.

The existing capacities and distribution systems at the Idaho National Engineering Laboratory for electricity, steam, water, and domestic wastewater treatment are adequate to support the receipt and storage of foreign research reactor spent nuclear fuel for all storage options.

Some of the electric power at the Idaho National Engineering Laboratory is generated onsite, and the remainder is provided by the Idaho Power Company. The Utah Power and Light Company Antelope Substation, which is located on the Idaho National Engineering Laboratory, connects to the Scoville Substation, from which electricity is distributed to various facilities over a 138-kilovolt loop at the Idaho National Engineering Laboratory.

All water supplies for the Idaho National Engineering Laboratory are obtained from the Snake River Plain aquifer through wells. Pumping totals approximately 7 million m^3 per year (1.8 billion gallons per year). ICPP has a coal-fired steam system. Natural gas is not used at the Idaho National Engineering Laboratory.

F.4.2.1.4 Waste Management

Waste production associated with the operation of the FAST and IFSF facilities is characteristic to wet and dry storage, respectively, and is discussed in detail in Sections F.4.2.2.1.13 and F.4.2.2.2.13.

F.4.2.1.5 Air Quality

Nonradiological Emissions: It is expected that the ambient concentration levels from incident-free operation of existing facilities would not change from baseline concentrations due to the addition of foreign research reactor spent nuclear fuel. The baseline ambient concentrations are given in Table F-41. They are all below applicable standards and guidelines.

Radiological Emissions: Radiological emissions from the receipt and storage of foreign research reactor spent nuclear fuel in the existing facilities at the Idaho National Engineering Laboratory are discussed in Section F.4.2.1.2.

F.4.2.1.6 Water Resources

The use of FAST and ISFS facilities for the interim storage of foreign research reactor spent nuclear fuel would not change the current levels of water and usage of these facilities. Nor would it change thermal discharges from cooling water or the quantity or quality of radioactive and nonradioactive wastewater effluents.

Table F-41 Maximum Impacts to Nonradiological Air Quality from Spent Nuclear Fuel[a,b] at Existing Facilities at the Idaho National Engineering Laboratory (Phase 1)

Pollutant	*Averaging Time*	*Applicable Standard ($\mu g/m^3$)[c]*	*Maximum Baseline Concentration ($\mu g/m^3$)*	*Baseline plus Foreign Research Reactor Spent Nuclear Fuel ($\mu g/m^3$)*	*Percent of Standard*
Criteria pollutants					
• Carbon Monoxide	1-hr	40,000	1,200	1,200	3.8
• Nitrogen Dioxide	Annual	100	14.1	14.1	14.1
• Lead	Quarterly	1.5	0.002	0.002	0.1
• Particulate Matter (PM_{10})	24-hr	150	112	112	75
	Annual	50	19	19	38
• Sulfur dioxide	3-hr	1,300	534	534	41.1
	24-hr	365	238	238	65.3
	Annual	80	4.2	4.2	5.3
Other pollutants mandated by Idaho					
• Total Suspended Particulates	24-hr	150	120[d]	120	80
	Annual	60	45	45	75
• Fluorides	Monthly	62,168	0	0	0
	Bimonthly	46,626	0	0	0
	Annual	31,084	0	0	0
Hazardous/toxic air pollutants (carcinogens)					
• Ammonia Hydroxide	8-hr	180	0.33	36	20
• Benzene	Annual	12	0.029	0.029	16
• Formaldehyde	Annual	770	0.012	0.012	16
• Hexone	8-hr	2,100	0	0	0
• Hydrofluoric Acid	8-hr	25	0	0	0
• Tributylphosphate	8-hr	25	0	0	0

[a] *Source: (DOE, 1995g).*

[b] *Listed concentrations are the maximum of those calculated at the Idaho National Engineering Laboratory site boundary, public access roads inside the Idaho National Engineering Laboratory site boundary, and the Craters of the Moon National Monument.*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283.*

[d] *The background concentration for the 24-hour standard is the same as the background for annual average concentration.*

Interim storage of foreign research reactor spent nuclear fuel in existing facilities would not affect the quality of water resources because it would be stored in contained storage pools or above-grade and below-grade dry storage containers isolated from the environment.

With respect to accident conditions, the Programmatic SNF&INEL Final EIS concluded that on the basis of a bounding accident scenario for high-level waste tank failure, accidental leakage would cause negligible impacts to water resources (DOE, 1995g).

F.4.2.2 New Facilities (Phase 2)

Analysis options 2B and 2C involve the use of new facilities. The environmental impacts analyzed relate to the construction and operation of these new facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities and energy; and waste management.

The impacts are presented in terms of storage technologies: dry storage in Sections F.4.2.2.1 and wet storage in Section F.4.2.2.2. Accident analysis, which is associated primarily with the storage technology rather than specific facilities, is presented in Section F.4.2.3.

F.4.2.2.1 Dry Storage

Analysis option 2B is associated with dry storage of foreign research reactor spent nuclear fuel in new facilities. This analysis option would require the construction of a new dry storage facility at the Idaho National Engineering Laboratory. The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 of this EIS and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis, the impacts from the larger dry vault design are presented.

F.4.2.2.1.1 Land Use

A new dry storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 15.4 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent approximately 2.6 percent and 0.1 percent of FY 1995 total expenditures for the Idaho National Engineering Laboratory. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be negligible. In addition, when compared to the projected FY 1995 work force at the Idaho National Engineering Laboratory of approximately 11,600 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to 8 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Idaho National Engineering Laboratory.

F.4.2.2.1.3 Cultural Resources

No direct impacts on any cultural resources would be expected from the construction and operation of a new dry storage facility. Surveys of previously disturbed areas at the Idaho National Engineering Laboratory found no eligible cultural resources. Native American treaty rights that would affect any future land use on the Idaho National Engineering Laboratory would not be impacted (DOE, 1995g). Because activities associated with spent nuclear fuel management would take place within existing facility areas currently engaged in similar activities, DOE does not expect any impacts to important Native American resources from alteration of the visual setting or noise associated with the construction or operation of any new facilities. DOE has developed plans to be in full compliance with cultural resource laws (DOE, 1995g).

F.4.2.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility would not adversely impact aesthetic or scenic resources. A new dry storage facility would not be visible from any onsite or offsite public access roads. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.2.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Idaho National Engineering Laboratory that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils, and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.2.2.1.6 Air Quality

Nonradiological Emissions: Potential impacts from construction activities at the Idaho National Engineering Laboratory would include fugitive dust from construction activities (e.g., clearing of land, grading, road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Construction of a new dry fuel storage facility would be located near the center of the Idaho National Engineering Laboratory. The construction of this facility would require disturbance of approximately 3.7 ha (9 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years). As outlined in Table F-42, the ambient air quality impacts associated with construction-related activities would be minimal and the Idaho National Engineering Laboratory compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-42 Estimated Maximum Concentrations of Criteria Pollutants at the Idaho National Engineering Laboratory Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*	*Construction Activities*
Idaho National Engineering Laboratory Boundary (μg/m³)[b]				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0274
	Annual	50	19	0.0014
• Carbon Monoxide	1-hr	40,000	1,200	2.42
	8-hr	10,000	340	0.97
• Sulfur Dioxide	3-hr	1,300	534	0.397
	24-hr	365	238	0.085
	Annual	80	4.2	0.004
• Nitrogen Dioxide	Annual	100	14.1	0.068
Public Roads Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.0050
	Annual	50	19	0.0006
• Carbon Monoxide	1-hr	40,000	1,200	6.69
	8-hr	10,000	340	1.28
• Sulfur Dioxide	3-hr	1,300	534	0.727
	24-hr	365	238	0.117
• Nitrogen Dioxide	Annual	100	14.1	0.211
Craters of the Moon Boundary (μg/m³)				
• Particulate Matter (PM_{10})	24-hr	150	112	0.00037
	Annual	50	19	0.00003
• Carbon Monoxide	1-hr	40,000	1,200	0.61
	8-hr	10,000	340	0.08
• Sulfur Dioxide	3-hr	1,300	534	0.054
	24-hr	365	238	0.009
	Annual	80	4.2	0.0006
• Nitrogen Dioxide	Annual	100	14.1	0.009

[a] *Source: DOE, 1995g.*

[b] *To convert to μg/ft³, multiply by 0.0283.*

No nonradiological air emissions would be expected during operation of a new dry storage facility. Any emissions would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions would be produced during construction of a new dry storage facility.

Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects, (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage, small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Section F.6. The radiological consequences of air emissions from dry storage operation at the Idaho National Engineering Laboratory are discussed in Section F.4.2.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.2.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 6,500 million l (1,717 million gal) for the Idaho National Engineering Laboratory, these amounts represent approximately a 0.03 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Idaho National Engineering Laboratory.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Idaho National Engineering Laboratory. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Idaho National Engineering Laboratory could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Idaho National Engineering Laboratory would still be well within the design capacities of treatment systems at the Idaho National Engineering Laboratory. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.2.2.1.8 Ecology

Terrestrial Resources: DOE expects that construction impacts, which would include the loss of some wildlife habitat due to land clearing and facility development, would be greatest under the Regionalization and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory. Construction impacts from foreign research reactor spent nuclear fuel storage would not be significant because the construction activity would take place either within the boundaries of heavily developed areas or adjacent to those areas. However, construction activities could provide opportunities for the spread of exotic plant species, such as cheatgrass and Russian thistle (DOE, 1995g).

Wetlands: There would be no construction impacts to wetlands, which would be excluded from development, and impacts to threatened and endangered species would be unlikely given the location of previously-developed areas and the maximum size of the affected area of 3.7 ha (9 acres). Construction activities at the Idaho National Engineering Laboratory probably would not affect either of the endangered species found onsite (e.g., bald eagle and peregrine falcon). Both of these birds of prey are associated with riparian areas, wetlands, and larger bodies of water (e.g., reservoirs) and inhabit dry upland areas only temporarily when migrating. Disturbance to other sensitive (but not Federally-listed) species (e.g., the burrowing owl, northern goshawk, ferruginous hawk, Swainson's hawk, gyrfalcon, Townsend's western big-eared bat, and pygmy rabbit) would be possible but unlikely given the scale of the planned construction. Any impacts would be negligible and would last only as long as construction activities continue (DOE, 1995g).

Threatened and Endangered Species: Representative impacts from operations would include the disturbance and displacement of animals (such as the pronghorn antelope) caused by the movement and noise of personnel, equipment, and vehicles. Such impacts would be greatest under the Regionalization by Fuel Type and Geography, and Centralization Alternatives under the Programmatic SNF&INEL Final EIS at the Idaho National Engineering Laboratory, which would involve a generally higher level of operational activity; however, these impacts would be minor (DOE, 1995g). DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the potential construction site of foreign research reactor spent nuclear fuel storage facilities at the Idaho National Engineering Laboratory, as required by the Endangered Species Act. Letters regarding consultation under the Endangered Species Act are included in Volume 2, Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995g).

F.4.2.2.1.9 Noise

Noise generated onsite by construction or operation of a new dry storage facility should not adversely affect the public or the Idaho National Engineering Laboratory environment. Noise generated by construction would be site-specific and short lived. A limited number of new construction jobs would be generated, but the resulting temporary increase in worker and truck traffic is expected to be insignificant within the context of existing site traffic loads. Noise generated by operation would not significantly impact the environment because the facility would be located adjacent to previously developed, industrialized areas. Rail shipments of foreign research reactor spent nuclear fuel would be a small fraction of the rail traffic on the Blackfoot-to-Arco Branch of the Union Pacific System line that crosses the Idaho National Engineering Laboratory. There may be a slight increase in truck traffic to and from the potential storage site, but it is not expected to result in a perceptible increase in traffic noise or any change in community reaction to noise along the major access routes to the Idaho National Engineering Laboratory (DOE, 1995g).

F.4.2.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Idaho National Engineering Laboratory. Again, traffic congestion would not be a significant problem.

Traffic due to operations of a new dry storage facility would not increase site levels because the required workers would be drawn from the existing Idaho National Engineering Laboratory labor force.

F.4.2.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-43 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-43 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Idaho National Engineering Laboratory (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Dry Storage Facility[a]	0.00056	2.8×10^{-10}	0.0045	0.0000023
Storage at: • New Dry Storage Facility	0	0	0	0

[a] *The doses for this new dry storage facility are assumed to be equal to those for IFSF/CPP-749.*

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the spent nuclear fuel from one facility to another, or preparing the spent nuclear fuel for shipment offsite. Analysis option 2B involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing dry and wet storage facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a dry storage facility at the end of Phase 1, and the receipt of 193 shipments of foreign research reactor spent nuclear fuel at the new dry storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. Collective doses were calculated for both dry storage designs, the vault and the dry cask. The assumptions and methodology used to calculate the doses are described in Section F.5 of this appendix.

Table F-44 presents the doses that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-44 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (New Dry Storage)

	Worker Population Dose (person-rem)		*Worker Population Risk (LCF)*	
	New Dry Storage Cask	*New Dry Storage Vault*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phases 1 and 2[a]	424	370	0.17	0.15
Phases 1 and 2[b]	416	363	0.17	0.15

[a] *Phase 1 at IFSF/CPP-749*

[b] *Phase 1 at FAST*

F.4.2.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-45. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-45 Annual Utility and Energy Requirements for New Dry Storage at the Idaho National Engineering Laboratory

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	208,000	800 - 1,000	0.48 percent
Fuel (l/yr)	11,123,400	0	0 percent
Water (l/yr)	6,500,000,000	1,590,000[a] 400,000[b]	0.025 percent[a] 0.006 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.1.13 Waste Management

Construction of a new dry storage facility at the Idaho National Engineering Laboratory would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-46. These quantities represent a very small percent increase above current levels at the Idaho National Engineering Laboratory. Existing waste management storage and disposal activities at the Idaho National Engineering Laboratory could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Idaho National Engineering Laboratory waste management capacities would be minimal.

Table F-46 Annual Waste Generated for New Dry Storage at the Idaho National Engineering Laboratory

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	22[a] 1[b]	0.5 percent[a] 0.02 percent[b]
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.29 percent[a] 0.074 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.2 Wet Storage

Analysis option 2C involves long-term wet storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory. This analysis option would require the construction of a new wet storage facility at the site (Implementation Alternative 5 of Management Alternative 1).

F.4.2.2.2.1 Land Use

A new wet storage facility could be located in one of several developed areas, including the ICPP. These areas, which have already been developed for industrial use, occupy about 4,560 ha (11,400 acres). Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. This represents about 0.06 percent of the developed space at these areas. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on the Idaho National Engineering Laboratory.

F.4.2.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 18.7 percent of the estimated FY 1995 total expenditures for the Idaho National Engineering Laboratory (600 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage.

These costs represent about 3.8 percent and 0.6 percent of FY 1995 total expenditures for the Idaho National Engineering Laboratory. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Idaho National Engineering Laboratory of approximately 11,600 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Idaho National Engineering Laboratory.

F.4.2.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.2.2.1.3).

F.4.2.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.2.2.1.4).

F.4.2.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.2.2.1.5).

F.4.2.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction, as presented in Section F.4.2.2.1.6.

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3-micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs.

For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission level for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE Order 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

No radiological emissions would be produced during construction of a new wet storage facility.

F.4.2.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (0.72 million gal), respectively. With an annual average water usage of approximately 6,500 million l (1,717 million gal) for the Idaho National Engineering Laboratory, these amounts represent an increase of about 0.03 percent and less than 0.04 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Idaho National Engineering Laboratory.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Idaho National Engineering Laboratory. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Idaho National Engineering Laboratory could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Idaho National Engineering Laboratory would still be well within the design capacities of treatment systems at the Idaho National Engineering Laboratory. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.2.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.2.2.1.8).

F.4.2.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.2.2.1.9).

F.4.2.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.2.2.1.10).

F.4.2.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities and storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-47 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Idaho National Engineering Laboratory for wet storage. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-47 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at • New Wet Storage Facility	0.00038	1.9×10^{-10}	0.0031	0.0000016
Storage at: • New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 2C involves the receipt of 644 shipments of foreign research reactor spent nuclear fuel into the existing facilities (IFSF/CPP-749 and FAST) during Phase 1, the preparation of 161 transportation casks for shipment to a wet storage facility at the end of Phase 1, and the receipt of 193 shipments directly from the ports into the new wet storage facility after Phase 1 operations. It was assumed that at the end of a 10-year period, the foreign research reactor spent nuclear fuel would have decayed sufficiently to be accommodated in larger capacity transportation casks, such as those currently used in the United States for commercial spent nuclear fuel. For the purpose of this analysis, the transportation casks used for intrasite shipping are assumed to have a capacity four times as large as the capacity of the transportation casks used for the marine transport of the foreign research reactor spent nuclear fuel to the United States. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-48 presents the population dose that would be received by the members of the working crew and the associated risk if that working crew handled the total number of transportation casks at the Idaho National Engineering Laboratory. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-48 Handling-Related Impacts to Workers at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

Facility	*Worker Population Dose (person-rem)*	*Worker Population Risk (LCF)*
Phase 1: IFSF/CPP-749	257	0.10
Phase 1 and Phase 2	367	0.15
Phase 1: FAST	250	0.10
Phase 1 and Phase 2	360	0.14

F.4.2.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Idaho National Engineering Laboratory would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-49. These requirements represent a small percent of current requirements for the Idaho National Engineering Laboratory. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity at the Idaho National Engineering Laboratory is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-49 Annual Utility and Energy Requirements for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	208,000	1,000 - 1,500	0.72 percent
Fuel (l/yr)	11,123,400	0	0 percent
Water (l/yr)	6,500,000,000	2,700,000[a] 1,500,000[b]	0.04 percent 0.02 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.2.2.2.13 Waste Management

Construction of a new wet storage facility at the Idaho National Engineering Laboratory would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-50. These quantities represent a very small percentage increase above current levels at the Idaho National Engineering Laboratory. Existing waste management storage and disposal activities at the Idaho National Engineering Laboratory could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing Idaho National Engineering Laboratory waste management capacities would be minimal.

Table F-50 Annual Waste Generated for New Wet Storage at the Idaho National Engineering Laboratory (Implementation Alternative 5 to Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	750	none	0 percent
Transuranic (m^3/yr)	712	none	0 percent
Solid Low-Level (m^3/yr)	4,795	16[a] 1[b]	0.33 percent 0.02 percent
Wastewater (l/yr)	540,000,000	1,590,000[a] 400,000[b]	0.3 percent 0.07 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.2.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Idaho National Engineering Laboratory is presented here based on the methodology presented in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-51 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-51 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	1.3	0.67	15	28
• Dropped Fuel Cask	0.0001	0.074	0.0033	0.83	0.12
• Aircraft Crash w\Fire	1×10^{-6}	180	2.9	2,000	120
Wet Storage Accidents[b]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Idaho National Engineering Laboratory. These annual risks are multiplied by the maximum duration of this implementation alternative to obtain conservative estimates of risks at the Idaho National Engineering Laboratory presented in Table F-52 .

Table F-52 Annual Risks of Accidents at Idaho National Engineering Laboratory

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.1×10^{-7}	5.5×10^{-8}	0.0012	0.0000018
• Dropped Fuel Cask	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	4.8×10^{-12}
• Aircraft Crash w\Fire	9.0×10^{-11}	1.5×10^{-12}	0.0000010	4.8×10^{-11}
Wet Storage Accidents[b]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *IFSF/CPP-749 or New Dry Storage Facility*

[b] *New Wet Storage and FAST Facility*

Table F-53 presents the frequency and consequences of the accidents analyzed for Idaho National Engineering Laboratory for new wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Idaho National Engineering Laboratory. These annual risks are multiplied by the maximum duration of implementation alternative at each site to obtain conservative estimates of risks at the Idaho National Engineering Laboratory. Table F-54 presents the risk estimates from this implementation alternative.

Table F-53 Frequency and Consequences of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Wet Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	0.0016	0.0036	0.43	0.14
• Accidental Criticality	0.0031	28	30	140	1800
• Aircraft Crash	1×10^{-6}	22	9.8	250	400

[a] *New Wet Storage Facility*

Table F-54 Annual Risks of Accidents at the Idaho National Engineering Laboratory (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Wet Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.3×10^{-10}	2.9×10^{-10}	0.000035	8.8×10^{-9}
• Accidental Criticality	4.4×10^{-8}	4.7×10^{-8}	0.00022	0.0000022
• Aircraft Crash	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	1.6×10^{-10}

[a] *New Wet Storage Facility*

F.4.2.3.1 Secondary Impact of Radiological Accidents at the Idaho National Engineering Laboratory

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.2.3. Radiological accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Idaho National Engineering Laboratory, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-51). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 3.1 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Idaho National Engineering Laboratory.

F.4.2.4 Cumulative Impacts at the Idaho National Engineering Laboratory

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions and current activities at the Idaho National Engineering Laboratory. The contemplated DOE actions are the proposed construction and operation of an accelerator facility for tritium production (along with associated support facilities) (DOE, 1995d), the management of DOE-owned spent nuclear fuel discussed in Appendix B of the Programmatic SNF&INEL Final EIS (DOE, 1995g), and the storage and disposition of weapons-usable fissile materials at the Idaho National Engineering Laboratory site.

Tables F-55 and F-55A summarize the cumulative impacts for land use, socioeconomics, nonradiological air quality, occupational and public health and safety, energy and water consumption, and waste generation. As shown in the tables, the contribution of foreign research reactor spent nuclear fuel management to the cumulative impacts at the Idaho National Engineering Laboratory would be minimal.

F.4.2.5 Unavoidable Adverse Environmental Impacts

The construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory would result in some adverse impacts to the environment. Changes in designs and other methods of mitigation could eliminate, avoid, or reduce most of these to minimal levels. The following paragraphs identify adverse impacts that mitigation could not reduce to minimal levels or avoid altogether.

The generation of some fugitive dust during construction would be unavoidable, but would be controlled by water and dust suppressants. Similarly, construction activities would result in some minor, yet unavoidable, noise impacts from heavy equipment, generators, and vehicles.

Table F-55 Cumulative Impacts at the Idaho National Engineering Laboratory

Environmental Impact Parameter	FRR SNF Contribution	Current Activities[a]	Other Activities[a]	Cumulative Impact
Land Use (acres)	9	11,400[b]	604	12,013[b]
Socioeconomics (persons)	190[c]/30[d]	(e)	1980[c]/1080[d]	2,170[c]/1,110[d]
Air Quality (nonradiological)	See Table F-55A	See Table F-55A	See Table F-55A	See Table F-55A
Occupational and Public Health and Safety				
• MEI Dose (rem/yr)	$5.6x10^{-7}$	0.000056	0.0000057	0.000062
LCF (per year)	$2.8x10^{-10}$	$2.8x10^{-8}$	$2.8x10^{-9}$	$3.1x10^{-8}$
• Population Dose (person-rem/yr)	0.0045	0.34	32	32.3
LCF (per year)	$2.25x10^{-6}$	0.00017	0.016	0.016
• Worker Collective dose (person-rem/yr)	10[f]	30	344	384
LCF (per year)	0.004	0.012	0.137	0.154
Energy and Water Consumption				
• Electricity (MW-hr/yr)	1,000	208,000	3,897,000[g]	4,106,000
• Fuel (million l/yr)	0	11.1	1.35	12.45
• Coal (tons/yr)	0	12,500	13,660	26,160
• Water (million l/yr)	2.2	6,500	1,314	7,816
Waste Generation				
• High-Level (m^3/yr)	0	750	160	910
• Low-Level (m^3/yr)	22	4,795	2,800	7,617
• Transuranic (m^3/yr)	0	712	46	758
• Mixed (m^3/yr)	0	243	8	251

FRR SNF = Foreign Research Reactor Spent Nuclear Fuel

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a tritium accelerator facility, and the disposition of weapons-usable fissile materials*

[b] *Two percent of the total Idaho National Engineering Laboratory site area of 570,000 acres*

[c] *Increase over baseline during construction activities*

[d] *Increase over baseline during operation activities*

[e] *Baseline working force is approximately 11,600 persons*

[f] *The dose is due to the handling of FRR SNF during receipt and transfer, averaged over 40 years*

[g] *Major portion is the requirement for electricity by the tritium production accelerator facility (3,740,000 MW-hr/yr)*

The maximum loss of habitat would involve the conversion of approximately 4 ha (10 acres) of previously disturbed habitat that is of low quality and limited use to wildlife.

The amount of radioactivity that incident-free operation of the spent nuclear fuel facilities would release is a small fraction of the cumulative operational releases at the Idaho National Engineering Laboratory and would be well below applicable regulatory standards.

F.4.2.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Idaho National Engineering Laboratory would consume irretrievable amounts of electrical energy,

Table F-55A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Idaho National Engineering Laboratory Boundary[a]

Pollutant	Averaging Time	Regulatory Standard ($\mu g/m^3$)	Cumulative Concentration ($\mu g/m^3$)[b]
Carbon Monoxide	1-hour	40,000	1,245 (3.1%)
	8-hour	10,000	354 (3.5%)
Nitrogen Oxides	Annual	100	15 (15%)
Surfur Dioxide	3-hour	1,300	660 (51%)
	24-hour	365	267 (73%)
	Annual	80	7.5 (9.3%)
Particulate Matter (PM_{10})	24-hour	150	82 (55%)
	Annual	50	5 (10%)

[a] *Concentrations represent: foreign research reactor spent nuclear fuel management, other DOE-owned spent fuel management, construction and operation of a tritium supply facility and recycling activities, storage and disposition of weapons-usable fissile material activities, and current activities*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

F.4.2.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Idaho National Engineering Laboratory could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Idaho National Engineering Laboratory would be minimal in most environmental media.

Pollution Prevention: DOE is committed to comply with Executive Order 12856, "Federal Compliance with Right-to-Know Laws and Pollution Prevention Requirements;" Executive Order 12873, "Federal Acquisition, Recycling and Waste Prevention;" and applicable DOE orders and guidance documents in planning and implementing pollution prevention at the Idaho National Engineering Laboratory. The DOE views source reduction as the first priority in its pollution prevention program, followed by an increased emphasis on recycling. Waste treatment and disposal are considered only when prevention or recycling is not possible or practical (DOE, 1995g).

Cultural Resources: The lack of detailed specifications associated with the potential construction at the Idaho National Engineering Laboratory under the various storage options prevents the identification of specific project impacts and mitigation measures for particular structures and facilities. Basic compliance under cultural resource law involves five steps that would be essentially the same under all alternatives. These steps are: (a) identification and evaluation of resources in danger of impact, (b) assessment of effects to these resources in consultation with the State Historic Preservation Office and representatives of the Shoshone-Bannock Tribes, (c) development of plans and documents to minimize any adverse effects, (d) consultation with the Advisory Council on Historic Preservation and Tribal representatives as to the appropriateness of mitigation measures, and (e) implementation of mitigation measures. Therefore, if a cultural resource survey has not been performed in an area planned for ground disturbance under one of the storage options, consultation would be initiated with the Idaho State Historic Preservation Office, and the survey would be conducted prior to any disturbance. If cultural resources were discovered, they would be evaluated according to National Register criteria. Wherever possible, important resources would be left undisturbed. If the impacts are determined to be adverse and it is not feasible to leave the resource undisturbed, then measures would be initiated to reduce impacts. All mitigation plans would be developed in consultation with the State Historic Preservation Office and the Advisory Council on Historic Preservation and would conform to appropriate standards and guidelines established for historic preservation activities by the Secretary of the Interior (DOE, 1995g).

Some actions may affect areas of religious, cultural, or historic value to Native Americans. DOE has implemented a Working Agreement to ensure communication with the Shoshone-Bannock Tribes, especially relating to the treatment of archaeological sites during excavation, as mandated by the Archaeological Resources Protection Act; the protection of human remains, as required under the Native American Graves Protection and Repatriation Act; and the free exercise of religion as protected by the American Indian Religious Freedom Act. In keeping with DOE Native American policy, DOE Order 1230.2, and procedures to be defined in the Final Cultural Resources Management Plan, DOE would conduct Native American consultation during the planning and implementation of the policy. Procedures for dealing with the inadvertent discovery of human remains would be consistent with the Native American Graves Protection and Repatriation Act. If human remains were discovered, DOE would notify all Tribes that have expressed an interest in the repatriation of graves as required under Native American Graves Protection and Repatriation Act, including the Shoshone-Bannock, Shoshone, Painte, and the Northwestern band of the Shoshone Nation. These Tribes would then have an opportunity to claim the remains and associated artifacts in accordance with the requirements of Native American Graves Protection and Repatriation Act (DOE, 1995g).

Traffic and Transportation: All onsite shipments of foreign research reactor spent nuclear fuel would be in compliance with ID Directive 5480.3, "Hazardous Materials Packaging and Transportation Safety Requirements." These requirements provide assurance that, under normal conditions, the Idaho National Engineering Laboratory would meet "as low as reasonably achievable" conditions, credible accident situations (those with probability of occurrence greater than 1 x 10^{-7} per year) would not result in a loss of shielding or containment or a criticality, and an unintentional release of radioactive material would generate a timely response (DOE, 1995g).

Accidents: The DOE would implement the Idaho National Engineering Laboratory emergency response programs, as appropriate, following the occurrence of an accident to prevent or mitigate consequences. These emergency response programs, implemented in accordance with 5500-DOE series orders, typically involve emergency planning, emergency preparedness, and emergency response actions. Participating government agencies with plans that are interrelated with the Idaho National Engineering Laboratory Emergency Plan for Action include: the State of Idaho, Bingham County, Bonneville County, Butte County, Clark County, Jefferson County, the Bureau of Indian Affairs, and Fort Hall Indian Reservation. When an emergency condition exists at a facility, the Emergency Action Director is responsible for recognition, classification, notification, and protective action recommendations. Each emergency response plan utilizes resources specifically dedicated to assist a facility in emergency management. These resources include, but are not limited, to the following (DOE, 1994h):

- Idaho National Engineering Laboratory Warning Communications Center,
- Idaho National Engineering Laboratory Fire Department,
- Facility Emergency Command Centers,

- DOE Emergency Operations Centers,
- County and State Emergency Command Centers,
- medical, health physics, and industrial hygiene specialists,
- protective clothing and equipment (respirators, breathing air supplies, etc.), and
- periodic training exercises and drills within and between the organizations involved in implementing the response plans.

F.4.3 Hanford Site

If the Hanford Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Hanford Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2) the Hanford Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Hanford Site until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Hanford Site under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly in Phase 2, the Hanford Site could receive TRIGA foreign research reactor spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative. As a Phase 2 site, the Hanford Site would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility constructed on the 200 Area Plateau or the FMEF, which is a partially completed, large, hot cell facility. The new dry storage facility is described in Section 2.6.5.1.1. Description of the FMEF is provided in Appendix F, Section F.3.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Hanford Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

3A. The spent nuclear fuel that was managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new dry storage facility constructed either at the 200 Area Plateau or at the FMEF. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed in the dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (22,700 elements). If the Hanford Site receives TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only Western spent nuclear fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Hanford Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Hanford Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and manage it in facilities sized accordingly. The impacts from the management of this lesser amount of spent nuclear fuel would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Hanford Site would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Hanford Site would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Hanford Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 3A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years, and therefore the amount of spent nuclear fuel available for acceptance would also be decreased. In this case, the Hanford Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Hanford Site would be bounded by analysis option 3A (above).

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and stored would remain constant. The impacts would be the same as in option 3A (above).

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent nuclear fuel in this case cannot be quantified; however, the upper limit, as considered under analysis option 3A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Hanford Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Hanford Site for Phase 2 until ultimate disposition. For this implementation alternative, an analysis option 3B, which is similar to 3A, is considered as follows:

3B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Hanford Site where it would be managed at a new wet storage facility constructed at either the 200 Area Plateau or the WNP-4 Spray Pond facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Hanford Site receives only TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory, or only western fuel, the dry storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Hanford Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Hanford Site is not considered.

F.4.3.1 Existing Facilities

Existing facilities at the Hanford Site include the FMEF and the WNP-4 Spray Cooling Pond for dry and wet storage, respectively, of foreign research reactor spent nuclear fuel. For this analysis, existing facilities at the Hanford Site were considered essentially as new because of the significant modifications that would be required to use them for foreign research reactor spent nuclear fuel storage. Handling and transfer operations at the FMEF and the WNP-4 Spray Cooling Pond would be used to support new dry and wet storage facilities, respectively. The evaluation of potential environmental impacts is presented in Section F.4.3.2 and reflects the foreign research reactor spent nuclear fuel storage options described in Section F.4.3.

F.4.3.2 New Facilities (Phase 2)

Analysis options 3A and 3B involve the use of new or major additions to existing facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.3.2.1 Dry Storage

Dry storage is associated with analysis option 3A, which would require the construction of a new dry storage facility near the 200 Area Plateau or at the FMEF (FMEF currently has handling and transfer, but not adequate storage capabilities). The dry storage option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and Appendix F, Section F.3. There are no environmental impact parameters that would discriminate between the two designs. For the purpose of this analysis the impacts from the larger dry vault design are presented.

F.4.3.2.1.1 Land Use

A new dry storage facility would be located in either the 200 Area Plateau or at the FMEF in the 400 Area. These areas have been generally developed for industrial use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land at either area. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be $370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $92.5 million. This represents approximately 7.2 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be $15.6 million for receipt and handling and $0.6 million for storage. These costs represent approximately 1.2 percent and 0.05 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct and secondary construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Hanford Site.

F.4.3.2.1.3 Cultural Resources

No direct impacts on any cultural resources in the 200 Area Plateau would be expected from construction or operation of the new dry storage facility. This site has been surveyed for cultural resources, and no prehistoric or historic archaeological properties were found. No indirect impacts would be anticipated because no known archaeological sites are present within approximately 4 km (2.5 mi) of the 200 Area Plateau. Because the site is in an industrialized area, construction would not alter the historic significance or association with the Manhattan Project and/or Cold War facilities located nearby.

No direct or indirect impacts are expected to any cultural resources of significance to the Yakama Indian Nation, the Confederated Tribes of the Umatilla Indian Reservation, or the Wanapum Band. This is based on the location of the 200 Area Plateau relative to sacred and culturally important areas which have been identified through ethno-historical research and interviews with elders of bands that formerly used the Hanford Site (DOE, 1995g).

Modification of FMEF for dry storage would be inside the fence of the 400 Area. No cultural resources are known to exist within that area. Because of its location, no cultural resources on the Hanford Site would be disturbed by construction.

F.4.3.2.1.4 Aesthetic and Scenic Resources

Any changes caused by construction and operation of either dry storage facility would be consistent with the existing overall visual environment of the Hanford Site. Topographic features would obstruct both candidate storage sites from the view of populated areas. Although the new dry storage facility could be seen from the farmland bluffs that overlook the Columbia River to the east, these lands are on private property that is not readily accessible to the public. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility (DOE, 1995g).

F.4.3.2.1.5 Geology

There are no unique geologic features or minerals of economic value on the Hanford Site that would be adversely impacted by site development. Construction of a new dry storage facility would result in localized impacts to surficial soils and would necessitate the clearing and grading of 3.7 ha (9 acres). Site preparation, land shaping, and grading activities associated with construction would present a slight to moderate erosion hazard, but would be controlled and minimized by implementing best-management practices. The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.3.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Emissions of sulfur dioxide and nitrogen dioxide would result entirely from diesel exhaust. For this analysis, all vehicular emissions were conservatively assumed to occur within 1 year during 200 ten-hour work days. As shown in Table F-56, air quality impacts associated with construction-related activities would be minimal, and compliance with Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population.

Table F-56 Estimated Maximum Concentrations of Criteria Pollutants at the Hanford Site Attributable to New Dry Storage Construction

Pollutant	Averaging Time	Ambient Standard[a]	Baseline Concentration[a]	Construction Activities
Hanford Site Boundary (μg/m³)				
• Total Suspended Particulate (TSP)	Annual	75	56	0.4
• Particulate Matter (PM_{10})	24-hr	150	81	14
• Particulate Matter (PM_{10})	Annual	50	27	0.4
Workplace (ppmv)				
• Sulfur Dioxide	Annual	52	0.5	0.4
• Nitrogen Dioxide	Annual	100	6,500	200

[a] *Source: DOE, 1995g*

Nonradiological emissions would not be expected during operation of a new dry storage facility for foreign research reactor spent nuclear fuel.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel would be expected. Based on fuel drying and storage operations conducted at the Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides, such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides that are released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of a new dry storage facility at the Hanford Site are discussed in Section F.4.3.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.

F.4.3.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent no more than a 0.04 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. The impact on water quality during operations would also be small. Existing water treatment facilities at the Hanford Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Hanford Site would still be well within the design capacities of treatment systems at the Hanford Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.3.2.1.8 Ecology

Terrestrial Resources: Vegetation within construction areas would be destroyed during land-clearing activities. Plant species that are dominant on the 200 Area Plateau include: big sagebrush, cheatgrass, and Sanberg's bluegrass. Total area destroyed would amount to about less than one percent of this community on the Hanford Site. Although the plant communities to be disturbed are well-represented on Hanford Site, they are relatively uncommon regionally because of the widespread conversion of shrub-steppe habitats to agriculture. Disturbed areas are generally recolonized by cheatgrass, a nonnative species, at the expense of native plants. Mitigation of these impacts would include minimizing the area of disturbance and revegetating with native species, including shrubs, and establishing a 3:1 acreage replacement habitat in concert with a habitat enhancement plan presently being developed for Hanford Site in general. Adverse impacts to vegetation on Hanford Site would be limited to the project area and vicinity, and would not affect the viability of any plant populations on the Hanford Site (Bergsman et al., 1994).

Construction of the new dry storage facility would have some adverse affect on animal populations. Less mobile animals, such as invertebrates, reptiles, and small animals within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to adjacent suitable habitat, and these individual animals might not survive and reproduce. Project facilities would displace about 3.7 ha (9 acres) of animal habitat for the life of the dry storage facility. Revegetated areas (e.g., construction laydown areas and buried pipeline routes) would be reinvaded by animal species from surrounding undisturbed habitats. The adverse impacts of construction are expected to be limited to the project area and vicinity and should not affect the viability of populations on the Hanford Site.

Very small quantities of radionuclides would be released to the atmosphere during dry storage facility operations. No organisms studied to date are reported to be more sensitive than man to radiation. Therefore, the effects of these releases on terrestrial organisms are expected to be minor (Bergsman et al., 1994).

Any impacts to the vegetation and animal communities would be mitigated by minimizing the amount of land disturbed during construction, employing soil erosion control measures during construction activities, and revegetating disturbed areas with native species. These mitigation measures would limit the amount of direct and indirect disturbance to the construction area and surrounding habitats and would speed the recovery process for disturbed lands (Bergsman et al., 1994).

Operational impacts on terrestrial biotic resources would include exposure of plants and animals to small amounts of radionuclides released during operation of the new dry storage facility. The levels of radionuclide exposure would be below those levels that produce adverse effects (Bergsman et al., 1994).

Wetlands: There are no wetlands on or near either candidate storage site (Bergsman et al., 1994).

Threatened and Endangered Species: Construction and operation of the new dry storage facility would remove 3.7 ha (9 acres) of relatively pristine big sagebrush/cheatgrass/Sanberg's bluegrass habitat. This sagebrush habitat is considered priority habitat by the State of Washington because of its relative scarcity in the State and its use as nesting/breeding habitat by loggerhead shrikes, sage sparrows, sage thrashers, burrowing owls, pygmy rabbits, and sagebrush voles (Bergsman et al., 1994).

Loggerhead shrikes, listed as a Federal candidate (Category 2) and State candidate species, forage on the proposed spent nuclear fuel site and are relatively common on the Hanford Site. This species is sagebrush-dependent, as it is known to select primarily tall big sagebrush as nest sites. Construction of the new dry storage facility would remove big sagebrush habitat which would preclude loggerhead shrikes from nesting there. Foreign research reactor spent nuclear fuel site development would also be expected to reduce the value of the site as foraging habitat for shrikes known to nest in adjacent areas (Bergsman et al., 1994).

Sage sparrows and sage thrashers, both State candidate species, occur in mature sagebrush/bunchgrass habitat at the Hanford Site. The sage sparrow was observed on the proposed site in a survey during spring 1994. These species are known to nest primarily in sagebrush. Construction of the new dry storage facility would preclude both of these species nesting there and reduce the site's suitability as foraging habitat for these species (Bergsman et al., 1994).

Dry storage facility construction is not expected to substantially decrease Hanford Site population of loggerhead shrikes, sage sparrows, or sage thrashers because similar sagebrush habitat is still relatively common on the Hanford Site. However, the cumulative effects of constructing the new dry storage facility, in addition to future developments that further reduce sagebrush habitat (causing further fragmentation of nesting habitat), could negatively affect the long-term viability of populations of these species on the Hanford Site (Bergsman et al., 1994).

Burrowing owls, a State candidate species, are relatively common on Hanford Site and nest in abandoned ground squirrel burrows on the 200 Area Plateau. Construction would remove sagebrush and disturb soil, displacing ground squirrels and thus reducing the suitability of the area for nesting by burrowing owls, and would also displace small mammals, which constitute a portion of the prey base for this species. Dry storage facility construction would not be expected to negatively impact the viability of the population of burrowing owls on the Hanford Site, as their use of ground squirrel burrows as nests is not limited to burrows in big sagebrush habitat (Bergsman et al., 1994).

Pygmy rabbits, a Federal candidate (Category 2) and State-listed threatened species, are known to utilize tall clumps of big sagebrush habitat throughout most of their range. However, this species has not recently been observed on the Hanford Site. Construction of the new dry storage facility would therefore reduce the potential for this species' occurrence by removing habitat suitable for its use (Bergsman et al., 1994).

Sagebrush voles, a State minor species, are common on Hanford Site and select burrow sites near sagebrush; however, this species is common only at higher elevations around the Hanford Site. Construction of the new dry storage facility would remove sagebrush habitat, precluding sagebrush voles from utilizing the site. However, construction would not affect the overall viability of sagebrush vole populations on Hanford Site because the majority of the population is found on the Fitzner/Eberhardt Arid Lands Ecology Preserve (Bergsman et al., 1994).

The closest known nests of ferruginous hawks, a Federal candidate (Category 2) and State threatened species, and Swainson's hawk, a State candidate, are 8.5 km (5 mi) and 6.2 km (3.7 mi), respectively, from the 200 Area Plateau. The potential site comprises a portion of the foraging range of these hawks. Construction of the new dry storage facility is not expected to disrupt the nesting activities of these species. However, construction would displace small mammal populations and thus reduce the prey for these birds. The cumulative effects of constructing the new dry storage facility, in addition to future reductions in sagebrush habitat (causing further fragmentation of foraging habitat), could negatively affect the long-term viability of populations of these two species on the Hanford Site (Bergsman et al., 1994).

Piper's daisy, listed as a State sensitive species, is relatively uncommon but widely distributed across the Hanford Site. Piper's daisy occurs in gravelly soils on the 200 Area Plateau. If construction of the new dry storage facility includes disturbing soils in the gravel pit, Piper's daisy would be eliminated in that area. However, because of the species' wide distribution, construction would not be expected to negatively affect the viability of this species on the Hanford Site (Bergsman et al., 1994).

DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction sites of foreign research reactor spent nuclear fuel storage facilities at the Hanford Site, as required by the Endangered Species Act.

The modification of FMEF for dry storage would take place within the fenced 400 Area. This area has already been disturbed and no further ecological impacts would be expected.

F.4.3.2.1.9 Noise

Noise generated onsite by construction and operation of a new dry storage facility should not adversely affect the public or the Hanford Site environment. Based on a noise impact analysis for locating a new production reactor at the Hanford Site, ambient noise levels would not exceed the limits set by Washington State or the Environmental Protection Agency. The analysis indicated that any increased traffic along the major roadways from construction and operation of the new production reactor would result in little or no increase in the annoyance level experienced by communities or individuals. As a result, no significant noise impacts from activities associated with the new dry storage facility construction and operation are expected at receptor locations outside the Hanford Site boundary or at residences along the major highways leading to either candidate storage site.

F.4.3.2.1.10 Traffic and Transportation

Construction materials, wastes, and excavated materials would be transported both onsite and offsite. These activities would result in increases in operation of personal-use vehicles by commuting construction workers, commercial truck traffic, and in traffic associated with the daily operations of the Hanford Site. Again, traffic congestion would not be a significant problem.

Traffic congestion, although moderate at shift changes, would not be noticeably worse due to this level of construction effort.

F.4.3.2.1.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Hanford Site would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-57 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-57 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• FMEF (dry storage)	0.00020	1.0×10^{-10}	0.011	0.0000055
• New Dry Storage Facility	0.00025	1.3×10^{-10}	0.015	0.0000075
Storage at:				
• FMEF (dry storage)	0	0	0	0
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 3A involves the receipt and unloading of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-58 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Hanford Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the

Table F-58 Handling-Related Impacts to Workers at the Hanford Site (New Dry Storage)

	Worker Population Dose (Person-rem)	Worker Population Risk (LCF)
	FMEF/New Dry Storage	FMEF/New Dry Storage
Phase 2	266/113[a]	0.11/0.05[a]

[a] *The two numbers represent the cask/vault designs respectively.*

administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Hanford Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-59. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-59 Annual Utility and Energy Requirements for New Dry Storage at the Hanford Site

Commodity	Baseline Site Usage	Dry Storage Usage	Percent Increase
Electricity (MW-hr/yr)	340,000	800 - 1,000	0.3 percent
Fuel (l/yr)	83,000,000	0	0 percent
Water (l/yr)	15,000,000,000	1,590,000[a] 400,000[b]	0.01 percent[a] 0.003 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.1.13 Waste Management

Construction of a new dry storage facility at the Hanford Site would generate 1,800 m^3 (2,340 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-60. These quantities, represent a very small percent increase above current levels at the Hanford Site. Existing waste management storage and disposal activities at Hanford Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Hanford Site waste management capacities would be minimal.

Table F-60 Annual Waste Generated for New Dry Storage at the Hanford Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	240	none	0 percent
Transuranic (m^3/yr)	170	none	0 percent
Solid Low-Level (m^3/yr)	20,000	22[a] 1[b]	0.11 percent[a] 0.005 percent[b]
Wastewater (l/yr)	210,000,000	1,590,000[a] 400,000[b]	0.75 percent[a] 0.2 percent[b]

[a] *During receipt and handling.*

[b] *During storage.*

F.4.3.2.2 Wet Storage

Analysis option 3B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Hanford Site. This storage option would require the construction of a new wet storage facility.

F.4.3.2.2.1 Land Use

A new wet storage facility would be located on the 200 Area Plateau or in conjunction with the WNP-4 Spray Cooling Pond. These areas have already been developed for industrial use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land at either area. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at either area would significantly impact land use patterns on the Hanford Site.

F.4.3.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be \$449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$112.2 million. This represents approximately 8.7 percent of the estimated FY 1995 total expenditures for the Hanford Site (1,288 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be \$23.3 million for receipt and handling and \$3.5 million for storage. These costs represent about 1.8 percent and 0.3 percent of FY 1995 total expenditures for the Hanford Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at Hanford Site of approximately 18,500 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be small to both the region of influence and the Hanford Site.

F.4.3.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.3.2.1.3).

The potential for impacting cultural resources would be even less for the WNP-4 Spray Pond because the structures are all essentially in place. Thus, there would be no opportunity for discovery of cultural resources during construction.

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.3.2.1.4).

F.4.3.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.3.2.1.5).

F.4.3.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of 2.8 ha (7 acres) of land. In comparison, 3.7 ha (9 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.3.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological-Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operation in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs as an indicator nuclide. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilation, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE 5480.1B (DOE, 1989b) for both onsite and offsite concentrations.

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the IFSF at the Idaho National Engineering Laboratory. The annual air emission for the IFSF was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas.

Radiological emissions from the operation of the new wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading and storage are provided in Section F.6.6.1.

No radiological emissions would be produced during construction of a new wet storage facility.

F.4.3.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (0.72 million gal), respectively. With an annual average water usage of approximately 15,000 million l (3,960 million gal) for the Hanford Site, these amounts represent an increase of about 0.02 percent and less than 0.005 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Hanford Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Hanford Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Hanford Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Hanford Site would still be well within the design capacities of treatment systems at the Hanford Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.3.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.3.2.1.8).

F.4.3.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.3.2.1.9).

F.4.3.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.3.2.1.10).

F.4.3.2.2.11 Occupational and Public Health and Safety

Emissions-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Hanford Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-61 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Hanford Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-61 Annual Public Impacts for Receipt and Storage of Foreign Research Reactor Spent Nuclear Fuel at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• WNP-4 Spray Pond	0.00022	1.1×10^{-10}	0.0058	0.0000029
• New Wet Storage Facility	0.00020	1.0×10^{-10}	0.012	0.000006
Storage at:				
• WNP-4 Spray Pond	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
• New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 3B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from Idaho National Engineering Laboratory and/or Savannah River Site and 193 shipments directly from the ports into a wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-62 presents the population dose that would be received by the members of the working crew and the associated risks if that working crew handled the total number of transportation casks at the Hanford Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

F.4.3.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Hanford Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-63. These requirements represent a small percent of current requirements for the Hanford Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Hanford Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-62 Handling-Related Impacts to Workers at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (person-rem)	Worker Population Risk (LCF)
	New Wet Storage Facility or WNP-4 Spray Pond	New Wet Storage Facility or WNP-4 Spray Pond
Phase 2	109	0.04

Table F-63 Annual Utility and Energy Requirements for New Wet Storage at the Hanford Site

Commodity	Baseline Site Usage	Wet Storage Usage	Percent Increase
Electricity (MW-hr/yr)	340,000	1,000-1,500	0.44 percent
Fuel (l/yr)	83,000,000	0	0 percent
Water (l/yr)	15,000,000,000	2,700,000[a] 1,500,000[b]	0.02 percent 0.01 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.3.2.2.13 Waste Management

Construction of a new wet storage facility at the Hanford Site would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-64. These quantities, represent a very small percentage increase above current levels at the Hanford Site. Existing waste management storage and disposal activities at Hanford Site could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing Hanford Site waste management capacities would be minimal.

Table F-64 Annual Waste Generated for New Wet Storage at the Hanford Site

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level Waste (m^3/yr)	240	none	0 percent
Transuranic Waste (m^3/yr)	170	none	0 percent
Solid Low-Level Waste (m^3/yr)	20,000	16[a] 1[b]	0.08 percent 0.005 percent
Wastewater (l/yr)	210,000,000	1,590,000[a] 400,000[b]	0.75 percent 0.2 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.3.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Hanford Site is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-65 presents frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-65 Frequency and Consequences of Accidents at the Hanford Site

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem)*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Fuel Assembly Breach	0.16	3.0	0.57	42	50
• Dropped Fuel Cask	0.0001	0.26	0.0085	3.0	0.22
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA
Dry Storage Accidents at FMEF					
• Spent Fuel Assembly Breach	0.16	4.7	2.1	46	0.99
• Dropped Fuel Cask	0.0001	0.2	0.032	3.2	0.0049
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA	NA

NA = Not Applicable

[a] *New Dry Storage Facility*

[b] *Aircraft Crash accidents are not applicable to Hanford Site because their frequency of occurrence is less than one every ten million years.*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Hanford Site. These annual risks are multiplied by the maximum duration of the implementation alternative at the Hanford Site to obtain conservative estimates of risks for the Hanford Site. These risk estimates are presented in Table F-66.

Table F-66 Annual Risks of Accidents at the Hanford Site

	Risks			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	2.4×10^{-7}	4.6×10^{-8}	0.0034	0.0000032
• Dropped Fuel Cask	1.3×10^{-11}	4.3×10^{-13}	1.5×10^{-7}	8.8×10^{-12}
• Aircraft Crash w\Fire[b]	NA	NA	NA	NA
Dry Storage Accidents at FMEF				
• Spent Nuclear Fuel Assembly Breach	3.7×10^{-7}	1.7×10^{-7}	0.0037	6.4×10^{-8}
• Dropped Fuel Cask	8 x 10-12	1.6×10^{-12}	1.6×10^{-7}	2.5×10^{-13}
• Aircraft Crash with Fire[b]	10^{-12}	NA	NA	NA

NA = Not Applicable

[a] *New Dry Storage Facility*

[b] *Aircraft crash accidents are not applicable to Hanford Site because their frequency of occurrence is less that one every ten million years*

Table F-67 presents the frequency and consequences of the accidents analyzed for the Hanford Site for new wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Hanford Site. These annual risks are multiplied by the

Table F-67 Frequency and Consequences of Accidents at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
New Wet Storage Facility:					
• Spent Nuclear Fuel Assembly Breach	0.16	0.13	0.0033	1.6	0.25
• Accidental Criticality	0.0031	64	14	740	3,600
• Aircraft Crash[a]	NA	NA	NA	NA	NA
WNP-4 Spray Pond:					
• Spent Nuclear Fuel Assembly Breach	0.16	0.15	0.0033	1.3	0.00024
• Accidental Criticality	0.0031	97	76	620	120
• Aircraft Crash[a]	NA	NA	NA	NA	NA

NA = Not Applicable

[a] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

maximum duration of this implementation alternative at the Hanford Site to obtain conservative estimates of risks at the Hanford Site. Table F-68 presents the risk estimates for this implementation alternative.

Table F-68 Annual Risks of Accidents at the Hanford Site (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
New Wet Storage Facility:				
• Fuel Assembly Breach	1.1×10^{-8}	2.7×10^{-10}	0.00013	1.6×10^{-8}
• Accidental Criticality	1.0×10^{-7}	2.2×10^{-8}	0.0012	0.0000044
• Aircraft Crash[a]	NA	NA	NA	NA
WNP-4 Spray Pond:				
• Fuel Assembly Breach	1.2×10^{-8}	2.7×10^{-10}	0.00011	1.5×10^{-11}
• Accidental Criticality	1.5×10^{-7}	1.2×10^{-7}	0.00096	1.5×10^{-7}
• Aircraft Crash[a]	NA	NA	NA	NA

NA = Not Applicable

[a] *Aircraft crash accidents are not applicable to the Hanford Site because their frequency of occurrence is less than one every ten million years.*

F.4.3.3.1 Secondary Impact of Radiological Accidents at the Hanford Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.3.3. Radiological accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident.

In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Hanford Site, the radiological accident with the highest MEI dose is the fuel assembly breach at a dry storage facility located at the FMEF (Table F-65). For this accident, the MEI dose would be 3.9 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage are expected at the Hanford Site.

F.4.3.4 Cumulative Impacts at the Hanford Site

This section presents the cumulative impacts of the proposed action, potential impacts of other major contemplated DOE actions, and current activities at the Hanford Site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g), the Management of Spent Nuclear Fuel from the K Basins Draft EIS (DOE, 1995d) and the Safe Interim Storage of Hanford Tank Wastes Final EIS (DOE, 1995c).

Table F-69 summarizes the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption and waste generation. The table also presents the contributions from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Hanford Site. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Hanford Site.

As shown in Table F-69, the contribution from management of foreign research reactor spent nuclear fuel to the cumulative impacts at the Hanford Site would be minimal. It is concluded, therefore, that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts.

F.4.3.5 Unavoidable Adverse Environmental Impacts

Unavoidable impacts associated with foreign research reactor spent nuclear fuel management activities would derive principally from construction activities needed for new storage facilities. There would be displacement of some animals from the construction site and the destruction of plant life within the area scoped for construction [up to 4 ha (10 acres)]. Criteria pollutants and radionuclides, would also be released in up to permitted quantities. Traffic congestion and noise would be expected to increase by a few percent during the construction of major facilities.

F.4.3.6 Irreversible and Irretrievable Commitments of Resources

The irreversible and irretrievable commitment of resources resulting from the construction and operation of facilities for the receipt and storage of foreign research reactor spent nuclear fuel would involve materials that could not be recovered or recycled or that would be consumed or reduced to unrecoverable forms. The construction and operation of facilities for foreign research reactor spent nuclear fuel facilities at the Hanford Site would consume irretrievable amounts of electrical energy, fuel, concrete, sand, and gravel. Other resources used in construction would probably not be recoverable. These would include

Table F-69 Cumulative Impacts at the Hanford Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	84,343[b]	84,352
Socioeconomics (persons)	190[c]/30[d]	3300/1220[e]	3,490[c]/1250[d]
Air Quality (nonradiological)	See Table F-56	NA	(f)
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	2.5×10^{-7}	0.0000036	0.0000036
LCF (per year)	1.3×10^{-10}	1.5×10^{-9}	1.5×10^{-9}
• Population dose (person-rem/yr)	0.015	0.22	0.235
LCF (per year)	0.0000075	0.00011	0.00011
• Worker Collective dose (person-rem/yr)	8.9[g]	116.5	125.4
LCF (per year)	0.0035	0.0466	0.05
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	495,600	496,600
• Fuel (million l/yr)	0	94.4	94.4
• Water (million l/yr)	2.2	15,004	15,006
Waste Generation			
• High-Level (m^3/yr)	0	354	354
• Low-Level (m^3/yr)	22	33,310	33,332
• Transuranic (m^3/yr)	0	240	240
• Mixed/hazardous (m^3/yr)	0	402	402

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of a Laser Interferometer Gravitational-Wave Observatory, decommissioning of unused facilities, site restoration activities interim storage and tank wastes, management of spent nuclear fuel from the K basins, and current activities.*

[b] *Current operational areas constitute 83,767 acres*

[c] *Increase over baseline, during construction activities*

[d] *Increase over baseline, during operation activities*

[e] *Current working force is approximately 18,500 persons*

[f] *Nonradiological ground level cumulative concentrations would be within regulatory standards. 24-hour concentration for fugitive dust may exceed limits during construction of more than one facility simultaneously.*

[g] *The dose is due to the handling of the Foreign Research Reactor Spent Nuclear Fuel during receipt averaged over 30 years*

finished steel, aluminum, copper, plastics, and lumber. Most of this material would be incorporated in foundations, structures, and machinery.

F.4.3.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that Hanford Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Hanford Site would be minimal in most environmental media.

Pollution Prevention: DOE is responding to Executive Order 12856 and associated DOE orders and guidelines by reducing the use of toxic chemicals; improving emergency planning, response, and accident notification; and encouraging the development and use of clean technologies and the testing of innovative pollution prevention technologies. Program components include waste minimization, source reduction and recycling, and procurement practices that preferentially procure products made from recycled materials. The pollution prevention program at the Hanford Site is being formalized in a Hanford Site Waste Minimization and Pollution Prevention Awareness Program Plan (DOE, 1995g).

The foreign research reactor spent nuclear fuel program activities would be conducted in accordance with this plan, and implementation of the pollution prevention and waste minimization plans would minimize impacts of wastes generated during spent nuclear fuel management activities (DOE, 1995g).

Socioeconomics: The level of predicted employment for foreign research reactor spent nuclear fuel activities at the Hanford Site is not large enough in comparison with present Hanford, local, or regional employment to produce a boom-bust impact on the economy (DOE, 1995g).

Cultural Resources: To avoid loss of cultural resources during construction of foreign research reactor spent nuclear fuel facilities on the Hanford Site, a cultural resources survey of the area of interest would be conducted by Pacific Northwest Laboratories Cultural Resources staff. Assuming no such resources were found, construction would proceed. If, however, during construction (earth moving) any cultural resource is discovered, construction activities would be halted and the Pacific Northwest Laboratories Cultural Resources staff called upon to evaluate and determine the appropriate disposition of the find.

To avoid loss of cultural resources during operation, such as unauthorized artifact collection, workers could be educated through programs and briefing sessions to inform them on applicable laws and regulations for site protection. These educational programs would stress the importance of preserving cultural resources and specifics of the laws and regulations for site protection. The exact locations of cultural resources are not identified by the Pacific Northwest Laboratories Cultural Resources group, therefore, any such artifact collection would be in an area discovered by the worker(s) (DOE, 1995g).

Geology: Soil loss would be controlled during construction using standard dust suppression techniques on disturbed soil and by stockpiling with cover where necessary. Following construction, soil loss would be controlled by revegetation and relandscaping of disturbed areas (DOE, 1995g).

Air Resources: To avoid impacts associated with emissions of fugitive dust during construction activities, exposed soils would be treated using standard dust suppression techniques. New facility sources of pollutant emissions to the atmosphere would be designed using best available technology to reduce emissions to "as low as reasonably achievable" levels (DOE, 1995g).

Water Resources: The impacts to surface and groundwater sources could be minimized through recycling of water, where feasible, and with cleanup of excess process water before release to ground or surface water (DOE, 1995g).

Noise: Generation of construction and operations noise would be reduced, as practicable, by using equipment that complies with noise guidelines (40 CFR Parts 201-211). Construction workers and other personnel working in environments exceeding U.S. Environmental Protection Agency-recommended guidelines during spent nuclear fuel storage, construction, or operation would be provided with earmuffs or earplugs approved by the Occupational Safety and Health Administration (29 CFR Part 1910). Because of the remote location of the Hanford Site foreign research reactor spent nuclear fuel activities, there would be no noise impacts with respect to the public for which mitigation would be necessary (DOE, 1995g).

Traffic and Transportation: At sites with increasing traffic concerns, DOE would encourage use of high-occupancy vehicles (such as vans or buses), implementing carpooling and ride-sharing programs, and staggering work hours to reduce peak traffic.

Occupational and Public Health and Safety: Although no radiological impacts on workers or the public were evident from the evaluation of incident-free foreign research reactor spent nuclear fuel activities at Hanford, further improvement in controls to protect both workers and the general public is a continuing activity. The "as low as reasonably achievable" principle would be used for controlling radiation exposure and exposure to hazardous/toxic substances. The Hanford Site would continue to refine its current emergency planning, emergency preparedness, and emergency response programs in place to protect both workers and the public (DOE, 1995g).

Site Utilities and Support Services: No mitigation measures beyond those identified for ground disturbance activities associated with bringing power and water to the foreign research reactor spent nuclear fuel site would appear necessary. In those cases, use of standard dust suppression techniques and revegetation of disturbed areas would mitigate ground disturbance impacts.

Accidents: The Hanford Site maintains an emergency response center and has emergency action plans and equipment to respond to accidents and other emergencies. These plans include training of workers, local emergency response agencies (such as fire departments) and the public communication systems and protocols, readiness drills, and mutual aid agreements. The plans would be updated to include consideration of new foreign research reactor spent nuclear fuel facilities and activities. Design of new facilities to current seismic and other facility protection standards would reduce the potential for accidents, and implementation of emergency response plans would substantially mitigate the potential for impacts in the event of an accident.

F.4.4 Oak Ridge Reservation

If the Oak Ridge Reservation site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Oak Ridge Reservation to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Oak Ridge Reservation would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Oak Ridge Reservation until ultimate disposition.

The amount of spent nuclear fuel that would be received and managed at the Oak Ridge Reservation under Management Alternative 1 is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, in Phase 2, the Oak Ridge Reservation could receive the aluminum-based foreign research reactor spent nuclear fuel managed at the Savannah River site during Phase 1, Eastern foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative.

As a Phase 2 site, the Oak Ridge Reservation would receive and manage foreign research reactor spent nuclear fuel at a new dry storage facility to be constructed on the West Bear Creek Valley Site. The location is preferred among the four locations considered in a siting study performed for spent nuclear fuel management (MMES, 1994). Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of environmental impacts from management of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation is based on the above considerations. The analysis options selected do not represent all possible combinations but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

4A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new dry storage facility until ultimate disposition. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed in the new dry storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States discussed in Section 2.2.2 introduce additional analysis options that could be considered for the Oak Ridge Reservation as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Oak Ridge Reservation would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for this amount of spent fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1b (Section 2.2.2.1), the Oak Ridge Reservation would receive only HEU from the Idaho National Engineering Laboratory and/or the Savannah River Site. The amount of spent nuclear fuel would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the management of this amount of fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Oak Ridge Reservation would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 4A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and, therefore, the amount of spent nuclear fuel available for acceptance would also be decreased. In this case, the Oak Ridge Reservation would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Oak Ridge Reservation would be bounded by analysis option 4A above.

- Under Implementation Subalternative 2b, (Section 2.2.2.2), the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in option 4A above.

- Under Implementation Alternative 3 (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. These arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of fuel, in this case, cannot be quantified, however, the upper limit, as considered under analysis option 4A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of the foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Oak Ridge Reservation.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Oak Ridge Reservation for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 4B, which is similar to 4A, is considered as follows:

4B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Oak Ridge Reservation where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel to be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

- Under Implementation Alternative 6 (Section 2.2.2.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Oak Ridge Reservation would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Oak Ridge Reservation is not considered.

F.4.4.1 Existing Facilities

There are no existing facilities for storing foreign research reactor spent nuclear fuel at Oak Ridge Reservation. Consequently, all potential environmental consequences from foreign research reactor spent nuclear fuel storage are related to new facility construction and operation.

F.4.4.2 New Facilities (Phase 2)

Analysis options 4A and 4B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.4.2.1.3 Cultural Resources

There are no known historical, archaeological, paleontological, or Native American traditional sites in or around the potential storage site. No impacts to cultural resources are expected from ground disturbance, noise, or air emissions during construction or operation of the facility. Consultation with the Tennessee State Historic Preservation Office prior to project implementation is required by Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend further studies of the potential storage site to verify that no archaeological areas would be disturbed by construction activities (DOE, 1995g).

F.4.4.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility for foreign research reactor spent nuclear fuel would have similar impact on aesthetic and scenic resources at the Oak Ridge Reservation as the construction of spent nuclear fuel facilities under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel facilities associated with the Centralization Alternative would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The maximum height of the buildings on the site would not exceed 12.8 m (42 ft) above ground level, or two to three stories. Since the buildings would be set into the south face of Pine Ridge, between Pine Ridge and Chestnut Ridge, the site would not be visible from areas outside the Reservation, with the possible exception of a limited section of Gallaher Road on the west side of the Clinch River, looking east along Bear Creek Valley and the Bear Creek Road which is accessible to the public. The site would be screened by appropriate vegetation so that the public views would not be affected. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.4.2.1.5 Geology

For the most part, geologic impacts from construction activities would be limited to soil disturbance; although in some areas, ripping or blasting of limestone, dolomite, or chert layers might be required. No extensive or unique geologic or mineral resources are found in or around the potential storage site, so no geological impacts would be expected (DOE, 1995g). The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

Because previously undisturbed areas would be used for new construction, some soil impacts from siting a new dry storage facility at the West Bear Creek Valley site would occur as a result of grading. Potential impacts from sediment runoff generated during construction activities would be minimized by implementation of soil erosion and sediment control measures. During operations, impacts to soil resources would be controlled by the planting or landscaping of land surfaces not covered by pavement and buildings (DOE, 1995g).

Major seismic activity and associated mass movement and subsidence are unlikely to occur during the construction or operation because faults in the area have not been active since the late Paleozoic Era (DOE, 1995g).

F.4.4.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts associated with construction include generation of fugitive dust (particulate matter) and smoke from earth moving and clearing operations and emissions from construction equipment. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed surfaces.

Construction of this facility would require the clearing of approximately 16 ha (40 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-70 presents the air quality impacts associated with the construction of a new dry storage facility at the Oak Ridge Reservation. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using the Environmental Protection Agency regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0, in conjunction with onsite meteorological data from 1991.

Table F-70 Estimated Maximum Concentrations of Criteria Pollutants at the Oak Ridge Reservation Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[a]	*Baseline Concentration*[b]	*Construction Activities*
Oak Ridge Reservation Boundary ($\mu g/m^3$)[c]				
• Particulate Matter (PM_{10})[c]	24-hr	150	84.9	0.5450
	Annual	50	0.43	0.0144
• Carbon Monoxide	1-hr	40,000	2,748.0	26.756
	8-hr	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hr	1,300	170.3	2.356
	24-hr	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxide	Annual	100	2.1	0.098

[a] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[b] *Source: DOE, 1995g*

[c] *To convert to $\mu g/ft^3$, multiply by 0.0283*

Nonradiological emissions are not expected during operation of a new dry storage facility.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel are expected. Based on fuel drying and storage operations conducted at Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides such as ^{58}Co, ^{60}Co, ^{59}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amounts of these radionuclides released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask, and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or in a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility were calculated based on the methodology and assumptions described in Appendix F, Section F.6. The radiological consequences of air emissions from the operation of the dry storage facilities at the Oak Ridge Reservation are discussed in Section F.4.4.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.4.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent no more than a 0.07 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Oak Ridge

Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.1.8 Ecology

Terrestrial Resources: Radiation doses received by terrestrial biota from foreign research reactor spent nuclear fuel activities would be expected to be similar to those received by man. Although guidelines have not been established for acceptance limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Evidence indicates that no other living organisms have been identified that are likely to be significantly more radiosensitive than man. Thus, so long as exposure limits protective of man were not exceeded, no significant radiological impact on populations of biota would be expected as a result of foreign research reactor spent nuclear fuel activities at the West Bear Creek Site (DOE, 1995g).

Under the Centralization Alternative, construction of the potential spent nuclear fuel management facility would result in the disturbance of approximately 36 ha (90 acres) [16 ha (40 acres) if foreign research reactor spent nuclear fuel is considered in isolation], or less than 1 percent of the Oak Ridge Reservation. It is assumed that the area to be disturbed includes construction laydown areas, grading, and new buildings, and that the access road or other rights-of-ways have not been included in the total area to be disturbed. Vegetation within the area of the potential site for the spent nuclear fuel management facility would be destroyed during land clearing activities, but may be mitigated by revegetating with native species where possible. Vegetation cover in this area is predominantly oak-hickory forest or pine-hardwood forest. Both forest types are common on the Oak Ridge Reservation and within the region (DOE, 1995g).

Construction of a new dry storage facility would have some adverse effects on animal populations. Less mobile animals, such as amphibians, reptiles, and small mammals, within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to nearby suitable habitat. The long-term survival of these animals would depend on whether the area to which they moved was at or below its carrying capacity. Areas that would be revegetated upon completion of construction would be of minimal value to most wildlife, but might be repopulated by more tolerant species (DOE, 1995g).

The Migratory Bird Treaty Act is primarily concerned with the destruction of migratory birds, as well as their eggs and nests. It could be necessary to survey construction sites for the nests of migratory birds prior to construction and/or avoid clearing operations during the breeding season (DOE, 1995g).

Activities associated with operation, such as noise, increased human presence and traffic, and night lighting could affect wildlife living immediately adjacent to the storage site. While these disturbances could cause some sensitive species to move from the area, most animals should be able to adjust (DOE, 1995g).

Wetlands: Construction of a new dry storage facility would likely displace the forested wetlands adjacent to tributaries of Grassy Creek flowing through the potential site. This unavoidable displacement of wetlands would be accomplished in accordance with U.S. Army Corps of Engineers and Tennessee Water Quality Control Administration requirements. The potential also exists to disturb wetlands further downstream through erosion and sedimentation. Such impacts would be controlled through implementation of a soil erosion and sediment control plan. Construction-related discharges to Grassy

Creek would be relatively low and have negligible impacts to wetlands associated with the creek. No impacts to wetlands are anticipated during facility operations (DOE, 1995g).

Construction of a new dry storage facility would require the rechanneling of tributaries to Grassy Creek that cross the potential site, thus causing the loss of this aquatic habitat. In addition, soil erosion due to construction could cause water quality changes (primarily sediment loading) to Grassy Creek and its tributaries. These impacts could be minimized by implementation of soil erosion and sediment control measures. No operational impacts to aquatic resources are anticipated. It is assumed that the potential project would have a water retention pond within the security fence that might provide minimal habitat for amphibians in the area.

Threatened and Endangered Species: No Federally-listed species are expected to be affected. Site surveys would be required to verify the presence of State-listed or other special status species. Land clearing activities could destroy protected plant species, such as purple fringeless orchid and pink lady's-slippers, that may occur within the site. State-listed species including the Cooper's, sharp-shinned, and red-shouldered hawks, the barn owl, and the black vulture, which potentially occur in the area, could be impacted by project activities. Approximately 16 ha (40 acres) of potential nesting and foraging habitat would be lost as a result of construction activities. Because this type of habitat is abundant in the area, the loss would not be expected to affect the viability of populations of these species. However, appropriate steps would be taken to prevent nest disturbance. DOE would consult with the Tennessee Department of Environment and Conservation as appropriate to avoid or mitigate imminent impacts to State-listed species (DOE, 1995g). DOE would also consult with the U.S. Fish & Wildlife Service regarding threatened and endangered species for the proposed construction sites of foreign research reactor spent nuclear fuel storage facilities at the Oak Ridge Reservation. Impacts to threatened and endangered species are not anticipated.

F.4.4.2.1.9 Noise

Noises generated on the Oak Ridge Reservation do not propagate offsite at levels that impact the general population. Thus, the Oak Ridge Reservation noise impacts for both the Centralization and Regionalization by Fuel Type and Geography Alternatives would be those resulting from transportation of personnel and materials to and from the site that affect nearby communities, and those resulting from onsite sources that may affect some wildlife near these sources (DOE, 1995g).

The transportation noises are a function of the size of the work force (e.g., an increased work force would result in increased employee traffic and corresponding increases in deliveries by construction crews). Such noise and activity associated with construction would be expected to have short-term effects on most wildlife. Under the Centralization Alternative, the projected Oak Ridge Reservation work force would increase by about nine percent in the years 2000 to 2002 during peak construction, and decrease thereafter. There would be a corresponding increase in private vehicle and truck trips to the site. The day-night average sound level at 15 m (50 ft) from the roads that provide access to the Oak Ridge Reservation would be expected to increase by less than 1 decibel. No change is expected in the community reaction to noise along these routes. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.4.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Oak Ridge Reservation and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operation traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 4,200 full time employees and about 409,500 people in the region of influence. Construction and operation employees would contribute little to the future traffic because they represent such a small percentage of the region of influence population growth (DOE, 1995g). This conclusion would also be valid for a new dry storage facility for foreign research reactor spent nuclear fuel.

F.4.4.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by the wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-71 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-71 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Oak Ridge Reservation (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• New Dry Storage Facility	0.089	4.5×10^{-8}	0.085	0.000043
Storage at:				
• New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 4A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-72 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Oak Ridge Reservation.

The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-72 Handling-Related Impacts to Workers at the Oak Ridge Reservation (New Dry Storage)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Dry Storage	*New Dry Storage*
Phase 2	266/113[a]	0.11/0.05[a]

[a] *The two numbers represent the cask/vault designs respectively*

F.4.4.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Oak Ridge Reservation would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-73. These requirements represent a small percent of current requirements for the Oak Ridge Reservation. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Oak Ridge Reservation is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-73 Annual Utility and Energy Requirements for New Dry Storage at the Oak Ridge Reservation

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	335,800	800 - 1,000	0.3 percent
Fuel (l/yr)	3,600[c]	0	0 percent
Water (l/yr)	3,060,000,000	1,590,000[a] 400,000[b]	0.05 percent[a] 0.01 percent[b]

[a] *During receipt and handling*

[b] *During storage*

[c] *Decatherms/yr of natural gas*

F.4.4.2.1.13 Waste Management

Construction of a new dry storage facility at the Oak Ridge Reservation would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in

Table F-74. These quantities represent a very small percent increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at Oak Ridge Reservation could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Oak Ridge Reservation waste management capacities would be minimal.

Table F-74 Annual Waste Generated for New Dry Storage at the Oak Ridge Reservation

Waste Form	Baseline Site Generation	Dry Storage Generation	Percent Increase
High-Level (m^3/yr)	0	0	0 percent
Transuranic (m^3/yr)	16	0	0 percent
Solid Low-Level (m^3/yr)	6,902	22[a] 1[b]	0.32 percent[a] 0.01 percent[b]
Wastewater (l/yr)	754,000,000	1,590,000[a] 400,000[b]	0.21 percent[a] 0.05 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.4.2.2 Wet Storage

Analysis option 4B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation. This storage option would require the construction of a new wet storage facility.

F.4.4.2.2.1 Land Use

A new wet storage facility would be located in a 36-ha (90-acres) area in the eastern portion of West Bear Creek Valley. The majority of the land in this area can be characterized as vacant, unused, and ready for development. Use of West Bear Creek Valley for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 16 ha (40 acres) of land. This represents about 44 percent of the space designated for foreign research reactor spent nuclear fuel storage; however, this represents only about 0.1 percent of the entire Oak Ridge Reservation. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on Oak Ridge Reservation.

F.4.4.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be $449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about $112.2 million. This represents approximately 8.2 percent of the estimated FY 1995 total expenditures for the Oak Ridge Reservation (1,174 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new wet storage facility are estimated to be $23.3 million for receipt and handling and $3.5 million for storage. These costs represent about 2 percent and 0.3 percent of FY 1995 total expenditures for the Oak Ridge Reservation. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Oak Ridge Reservation of approximately 17,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Oak Ridge Reservation.

F.4.4.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.4.2.1.3).

F.4.4.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.4.2.1.4).

F.4.4.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.4.2.1.5).

F.4.4.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction (Section F.4.4.2.1.6).

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double banked high efficiency particulate air filters prior to release to the environment through an exhaust air system. The high efficiency particulate air filters would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility's exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b). Air emissions from the wet storage facility are expected to be similar to the air emissions from the CPP-603 at the Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Section F.6.

F.4.4.2.2.7 Water Resources

The annual water usage during construction and operations of a new wet storage facility is estimated to be about 1.9 million l (502,000 gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 3,060 million l (808 million gal) for the Oak Ridge Reservation, these amounts represent an increase of about 0.06 percent and 0.09 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Oak Ridge Reservation.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Oak Ridge Reservation. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Oak Ridge Reservation could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Oak Ridge Reservation would still be well within the design capacities of treatment systems at the Oak Ridge Reservation. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.4.2.2.8 Ecology

Impacts to ecology would be the same as for new dry storage (Section F.4.4.2.1.8).

F.4.4.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.4.2.1.9).

F.4.4.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.4.2.1.10).

F.4.4.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Oak Ridge Reservation would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from routine airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-75 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Oak Ridge Reservation. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-75 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI DOSE (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Wet Storage Facility	0.060	3.0×10^{-8}	0.061	0.000031
Storage at: • New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 4B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site, and 193 shipments directly from ports into a wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-76 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Oak Ridge Reservation. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-76 Handling-Related Impacts to Workers at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Wet Storage	*New Wet Storage*
Phase 2	109	0.04

F.4.4.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at Oak Ridge Reservation would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in

Table F-77. These requirements represent a small percent of current requirements for the Oak Ridge Reservation. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Oak Ridge Reservation is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-77 Annual Utility and Energy Requirements for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Commodity	Baseline Site Usage	Wet Storage Usage	Percent Increase
Electricity (MW-hr/yr)	335,800	800 - 1,000	0.15 percent
Fuel (l/yr)	3,600[a]	0	0 percent
Water (l/yr)	3,060,000,000	2,700,000[b] 1,500,000[c]	0.09 percent 0.05 percent

[a] *Decatherms/yr of natural gas*

[b] *During receipt and handling*

[c] *During storage*

F.4.4.2.2.13 Waste Management

Construction of a new wet storage facility at the Oak Ridge Reservation would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-78. These quantities represent a very small percentage increase above current levels at the Oak Ridge Reservation. Existing waste management storage and disposal activities at the Oak Ridge Reservation could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on existing the Oak Ridge Reservation waste management capacities would be minimal.

Table F-78 Annual Waste Generated for Wet Storage at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

Waste Form	Baseline Site Generation	Wet Storage Generation	Percent Increase
High-Level (m^3/yr)	0	0	0 percent
Transuranic (m^3/yr)	16	0	0 percent
Solid Low-Level (m^3/yr)	6,902	16[a] 1[b]	0.23 percent 0.01 percent
Wastewater (l/yr)	754,000,000	1,590,000[a] 400,000[b]	0.21 percent 0.05 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.4.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Oak Ridge Reservation is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiation doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-79 presents the frequencies and the consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-79 Frequency and Consequences of Accidents at the Oak Ridge Reservation

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	22	42	55	140
• Dropped Spent Nuclear Fuel Cask	0.0001	1.4	0.18	15	0.61
• Aircraft Crash w\Fire	0.000001	2300	180	2900	610

[a] *New Dry Storage Facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks for the Oak Ridge Reservation. These risk estimates are presented in Table F-80.

Table F-80 Annual Risks of Accidents at the Oak Ridge Reservation

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	0.0000018	0.0000034	0.0044	0.0000088
• Dropped Spent Nuclear Fuel Cask	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	2.4×10^{-11}
• Aircraft Crash w/Fire	1.2×10^{-9}	9.0×10^{-11}	0.0000015	2.4×10^{-10}

[a] *New Dry Storage Facility*

Table F-81 presents the frequency and consequences of the accidents analyzed for each site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Oak Ridge Reservation. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks at the Oak Ridge Reservation. Table F-82 presents the risk estimates from this implementation alternative.

F.4.4.3.1 Secondary Impact of Radiological Accidents at the Oak Ridge Reservation

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.4.3. Radiological

Table F-81 Frequency and Consequences of Accidents at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
New Wet Storage Facility					
• Spent Nuclear Fuel Assembly Breach	0.16	0.71	0.20	16	0.68
• Accidental Criticality	0.0031	1,500	3,300	1,400	6,800
• Aircraft Crash	0.000001	380	600	2,900	1,900

Table F-82 Annual Risks of Accidents at the Oak Ridge Reservation (Implementation Alternative 5 of Management Alternative 1)

	Risks			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
New Wet Storage Facility				
• Spent Nuclear Fuel Assembly Breach	5.5×10^{-8}	1.6×10^{-8}	0.0013	4.4×10^{-8}
• Accidental Criticality	0.0000024	0.000005	0.0022	0.0000084
• Aircraft Crash	1.9×10^{-10}	3.0×10^{-10}	0.0000015	7.6×10^{-10}

accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Oak Ridge Reservation, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire. For this accident, the MEI dose would be 2,300 mrem. For the air immersion and ground surface pathways only, the dose would be 140 mrem, which is greater than the 100 mrem limit used in this analysis. Local contamination would be likely around the dry storage facility, but is expected to be contained entirely within the boundaries of the Oak Ridge Reservation. Cleanup activities should be small and any impacts to land uses, cultural resources, water quality, and ecology would be reversible. No impacts to national defense or local economies would be expected.

F.4.4.4 Cumulative Impacts at the Oak Ridge Reservation

This section presents the cumulative impacts of the proposed action, potential impacts of other contemplated major DOE actions, and current activities at the site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g), the Tritium Supply and Recycling Final EIS (DOE, 1995a), and the Disposition of Surplus Highly Enriched Uranium Draft EIS (DOE, 1995e). Other activities considered for the Oak Ridge Reservation which could affect the site environment have not been determined sufficiently at this time to allow impact evaluation. They include activities associated with the waste management at the site, storage and disposition of weapons-usable fissile materials, and stockpile stewardship and management program.

Tables F-83 and F-83A summarize the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption, and waste generation at the site. Table F-83 also presents the contribution from the storage of foreign research reactor spent nuclear fuel on the cumulative impacts at the Oak Ridge Reservation. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Oak Ridge Reservation.

As shown in Table F-83, the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts (under the Centralization Alternative) at the Oak Ridge Reservation would be minimal. The Programmatic SNF&INEL Final EIS concludes that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts (DOE, 1995g). This conclusion is also valid for the implementation of any of the alternatives considered in this EIS for storage of foreign research reactor spent nuclear fuel at the Oak Ridge Reservation.

F.4.4.5 Unavoidable Adverse Environmental Impacts

Construction of the potential foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 16 ha (40 acres) of mostly forested undeveloped land. Although this represents less than one percent of the undeveloped land on the Oak Ridge Reservation, it would eliminate potential foraging and nesting habitat and would destroy plant species in the area. It would also require the dedication of a reasonably level land parcel that could otherwise accommodate other construction projects.

F.4.4.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel storage facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Most of the water that would be withdrawn from the Clinch River to operate the foreign research reactor spent nuclear fuel facilities would be returned to surface water in the Clinch River watershed, although some evaporative losses would be unavoidable. The land dedicated to the foreign research reactor spent nuclear fuel facilities could become available for other urban uses following closure and decommissioning. However, the soils on the site would have to be amended to support land uses such as agriculture, forestry, or wildlife management.

F.4.4.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Oak Ridge Reservation could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the Oak Ridge Reservation would be minimal in most environmental media.

Pollution Prevention: The DOE Oak Ridge Field Office established a Waste Minimization and Pollution Prevention Awareness Plan to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at the Oak Ridge Reservation. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the public. All contractors and users that exceed the U.S. Environmental Protection Agency criteria for small-quantity generators are

Table F-83 Cumulative Impacts at the Oak Ridge Reservation

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	40	14,335[b]	14,375
Socioeconomics (persons)	190[b]/30[c]	3,917[b]/930[c]	4,107[b]/960[c]
Air Quality (nonradiological)	See Table F-83A	See Table F-83A	See Table F-83A
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	0.00009	0.0155	0.0156
LCF (per year)	$4.5x10^{-8}$	0.0000077	0.0000078
• Population Dose (person-rem/yr)	0.085	94.5	94.6
LCF (per year)	0.000043	0.047	0.047
• Worker Collective Dose (person-rem/yr)	8.9[d]	261.3	270.2
LCF (per year)	0.0036	0.104	0.108
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	4,981,000[e]	4,982,000
• Natural Gas (million m^3/yr)	0	68.64	68.64
• Coal (tons/yr)	0	35,053	35,053
• Diesel Oil (million l/yr)	0	4.83	4.83
• Water (million l/yr)	2.2	68,172	68,174
Waste Generation			
• High-Level (m^3/yr)	0	0	0
• Low-Level (m^3/yr)	22	34,989	35,011
• Transuranic (m^3/yr)	0	16	16
• Mixed/Hazardous (m^3/yr)	0	119,411	119,411

[a] *Other activities include: DOE-owned spent nuclear fuel management, construction and operation of the Expended Core Facility, the construction and operation of the Advanced Neutron Source Facility, construction and operation of a Tritium production facility, and surplus highly enriched uranium management activities at the site*

[b] *Increase over baseline (17,000), during construction activities*

[c] *Increase over baseline (17,000), during operation activities*

[d] *The dose is due to the handling of Foreign Research Reactor Spent Nuclear Fuel during receipt averaged over 30 years*

[e] *Major portion of the requirement for electricity by the proposed tritium production facility (3,740,000 MW-hr/yr)*

establishing their own waste minimization and pollution prevention awareness programs. Contractor programs ensure that waste minimization activities are in accordance with Federal, State, and local environmental laws and regulations, and DOE orders (DOE, 1995g).

Additional goals include the promotion and use of nonhazardous materials, establishment of a baseline of waste generation data, calculations of annual reductions of waste generated, and implementation of recycling programs. Goals also include incorporation of waste minimization concepts and technologies in planning and design of new processes and facilities, and in upgrades of existing facilities. A waste minimization task force composed of representatives from each contractor has been established to coordinate waste minimization and pollution awareness activities (DOE, 1995g).

Socioeconomics: To reduce construction- and operation-related impacts, coordination with local communities could address potential impacts from increased labor and capital requirements. The knowledge of the extent and effect of growth due to foreign research reactor spent nuclear fuel

Table F-83A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Oak Ridge Reservation[a]

Pollutant	Averaging Time	Regulatory Standard (μg/m³)	Cumulative Concentration[b] (μg/m³)
Carbon Monoxide	1-hour	40,000	3,696 (9.2%)
	8-hour	10,000	2,495 (24.9%)
Nitrogen Oxides	Annual	100	13 (13%)
Surfur Dioxide	3-hour	1,300	336.6 (25.9%)
	24-hour	365	5.84 (1.6%)
	Annual	80	3.62 (4.52%)
Particulate Matter (PM_{10})	24-hour	150	88.1 (58.7%)
	Annual	50	0.48 (0.96%)
Total Suspended Particulates	Annual	150	119 (79.3%)

[a] *Concentrations represent activities from: foreign research reactor spent nuclear fuel management, DOE-owned spent fuel management, construction and operation of the Expended Core Facility, construction and operation of the Advanced Neutron Source Facility, construction and operation of a tritium supply and recycling facility, and surplus highly enriched uranium management at the site*

[b] *Numbers in parentheses indicate the percentage of the regulatory standard*

management activities could greatly enhance the ability of affected jurisdictions to plan effectively. Effective planning would address change in levels of service for housing, infrastructure, utilities, transportation, and public services and finances (DOE, 1995g).

To alleviate potential impacts associated with the in-migration of labor, local labor force availability could be increased through various employment training and referral systems currently provided by the Oak Ridge Reservation. The goal of these systems would be to reduce the potential for in-migration of labor to support foreign research reactor spent nuclear fuel management activities (DOE, 1995g).

Water Resources: The potential foreign research reactor spent nuclear fuel storage facilities would have to be located and constructed to minimize floodplain impacts and to avoid floodplains to the maximum extent possible, as required by Executive Order 11988 (Floodplain Management) and DOE rule 10 CFR 1022. Site-specific surveys would be performed to determine locations of flooding elevations more accurately (DOE, 1995g).

Ecology: DOE would consult with the Tennessee Department of Environment and Conservation as appropriate to avoid or mitigate imminent impacts to State-listed species (DOE, 1995g).

Accidents: New foreign research reactor spent nuclear fuel storage facilities would be designed to comply with current Federal, State, and local laws, DOE orders, and industrial codes and standards. This would provide facilities that are highly resistant to the effects of severe natural phenomena, including earthquakes, floods, tornadoes, high winds, as well as credible events as appropriate to the site, such as fires and explosions, and manmade threats to its continuing structural integrity for containing materials (DOE, 1995g).

Emergency preparedness plans have also been prepared for existing facilities and would be revised for new facilities to lower the potential consequences of an accident to workers and the public. All workers receive evacuation training to ensure timely and orderly personnel movement away from high-risk areas. Plans and arrangements with local authorities would also be in place to evacuate the general public that may be at risk of exposure to hazardous materials that are accidentally released (DOE, 1995g).

F.4.5 Nevada Test Site

If the Nevada Test Site is the site to manage DOE-owned spent nuclear fuel under the Programmatic SNF&INEL Final EIS, foreign research reactor spent nuclear fuel would be received and managed first at the Savannah River Site and/or the Idaho National Engineering Laboratory for the period required for the Nevada Test Site to construct and to place in operation new facilities to accommodate the spent nuclear fuel. As discussed in previous sections, this period (Phase 1) is estimated to be about 10 years. At the end of Phase 1 (e.g., start of Phase 2), the Nevada Test Site would be able to receive and manage foreign research reactor spent nuclear fuel that would be shipped from the Savannah River Site and/or the Idaho National Engineering Laboratory, and directly from the ports for those shipments made after Phase 1 concludes. Management of the foreign research reactor spent nuclear fuel would continue at the Nevada Test Site until ultimate disposition.

Although the Nevada Test Site has no existing facilities to receive foreign research reactor spent nuclear fuel at the beginning of the policy period, it has facilities that could be modified to receive foreign research reactor spent nuclear fuel within 5 years. These facilities are large hot cells located in the Nevada Research and Development Area on Jackass Flats. Presently these facilities (e.g., E-MAD) have little usage, but some are in acceptable condition. To use the E-MAD facility, a small pool would have to be constructed to be used for transferring the spent nuclear fuel from the transportation casks to containers designed for dry storage. A description of the E-MAD facility is included in Appendix F (Section F.3). The E-MAD facility could be ready within 5 years of the start of the proposed policy period.

The amount of spent nuclear fuel that would be received and managed at the Nevada Test Site under Management Alternative 1, is dictated by the distribution considered in the Programmatic SNF&INEL Final EIS. Accordingly, during Phase 2, the Nevada Test Site could receive the TRIGA spent nuclear fuel managed at the Idaho National Engineering Laboratory during Phase 1, Western foreign research reactor spent nuclear fuel under the Regionalization by Geography Alternative, or all foreign research reactor spent nuclear fuel under the Centralization Alternative.

As a Phase 2 site, the Nevada Test Site would receive and manage foreign research reactor spent nuclear fuel at a newly constructed dry storage facility or a modified E-MAD facility. Description of the new dry storage facility is provided in Section 2.6.5.1.1.

The analysis of potential environmental impacts from management of foreign research reactor spent nuclear fuel at the Nevada Test Site is based on the above considerations. The analysis options selected do not represent all possible combinations, but a reasonable set that provides a typical, and in some cases, bounding estimate of the resulting impacts.

The specific analysis options are as follows:

5A. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site, where it would be managed at a new dry storage facility or a modified E-MAD facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new or E-MAD facility until ultimate disposition. For the purposes of this analysis, the total amount of spent nuclear fuel that would be managed would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements).

The implementation alternatives of Management Alternative 1 for managing foreign research reactor spent nuclear fuel in the United States, discussed in Section 2.2.2, introduce additional analysis options that could be considered for the Nevada Test Site as follows:

- Under Implementation Subalternative 1a (Section 2.2.2.1), the amount of spent nuclear fuel to be received in the United States would be reduced to 5,000 elements. In this case, the Nevada Test Site would receive the foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and manage it in facilities sized for the reduced amount of spent nuclear fuel. The impacts from the management of this amount of spent nuclear fuel would be bounded by analysis option 5A above.

- Under Implementation Subalternative 1b (Section 2.3.1), the Nevada Test Site would receive from the Idaho National Engineering Laboratory and/or the Savannah River Site only HEU. The amount of HEU would be approximately 4.6 MTHM, representing 11,200 elements. The impacts from the storage of this amount of fuel would be bounded by analysis option 5A (above).

- Under Implementation Subalternative 1c (Section 2.2.2.1), the Nevada Test Site would receive target material in addition to the foreign research reactor spent nuclear fuel considered under the basic implementation of Management Alternative 1. The receipt and management of this material, which represents in uranium content approximately 620 typical foreign research reactor spent nuclear fuel elements, would increase the impacts of analysis option 5A by a small percentage.

- Under Implementation Subalternative 2a (Section 2.2.2.2), the duration of the policy would be decreased to 5 years and, therefore, the amount of spent nuclear fuel available for acceptance would also be decreased. In such a case, the Nevada Test Site would receive all foreign research reactor spent nuclear fuel from the Savannah River Site and/or the Idaho National Engineering Laboratory. The impacts from the management of the decreased amount of spent nuclear fuel at the Nevada Test Site would be bounded by analysis option 5A above.

- Under Implementation Subalternative 2b (Section 2.2.2.2) the acceptance of a small portion of the spent nuclear fuel would be extended over an indefinite period of time, but the amount of spent nuclear fuel to be received and managed would remain constant. The impacts would be the same as in analysis option 5A.

- Under Implementation Subalternative 3, (Section 2.2.2.3), DOE and the Department of State would consider alternative financial arrangements. The various arrangements would affect the amount of spent nuclear fuel that would be accepted by the United States as the foreign research reactor operators would consider their own alternatives on whether to send the spent nuclear fuel to the United States. The amount of spent fuel, in this case, cannot be quantified; however, the upper limit, as considered under analysis option 5A, would be bounding.

- Under Implementation Alternative 4 (Section 2.2.2.4), DOE and the Department of State would consider alternatives for the location where title of foreign research reactor spent nuclear fuel would be taken. The choices do not affect the management impacts at the Nevada Test Site.

- Under Implementation Alternative 5 (Section 2.2.2.5), DOE would consider construction of a new wet storage facility at the Nevada Test Site for Phase 2 until ultimate disposition. For this implementation alternative an analysis option 5B, which is similar to 5A, is considered as follows:

5B. The spent nuclear fuel managed at the Idaho National Engineering Laboratory and/or the Savannah River Site during Phase 1 would be shipped to the Nevada Test Site where it would be managed at a new wet storage facility. Spent nuclear fuel arriving in the United States after Phase 1 concludes would also be received and managed at the new facility until ultimate disposition. For the purposes of analysis, the total amount of spent nuclear fuel that would be managed in the wet storage facility would be all the foreign research reactor spent nuclear fuel eligible under the policy (approximately 22,700 elements). If the Nevada Test Site receives TRIGA spent nuclear fuel from the Idaho National Engineering Laboratory or only western spent fuel, the wet storage facility would be sized accordingly. The impacts from a smaller size facility would be bounded by the option analyzed.

- Under Implementation Alternative 6 (Section 2.3.6), DOE and the Department of State would consider chemical separation of foreign research reactor spent nuclear fuel in the United States. Based on the discussion in Section 2.3.6, the Nevada Test Site would not be considered as a site for chemical separation.

Under Management Alternative 3 (Hybrid Alternative) the Nevada Test Site is not considered.

F.4.5.1 Existing Facilities

Existing facilities considered for foreign research reactor spent nuclear fuel storage at the Nevada Test Site include the E-MAD facility in Area 25. For this analysis, the E-MAD facility was considered essentially as new because of the significant modifications needed to use it for foreign research reactor spent nuclear fuel storage. These modifications could be completed sometime between 1996 and 2006. The potential environmental impacts associated with the modification would be bounded by the impacts associated with the construction of a dry storage facility presented in Section F.4.5.2. Impacts from the operation of the E-MAD facility are presented below.

F.4.5.1.1 Socioeconomics

Potential socioeconomic impacts associated with storage option 5A would be attributable to staffing requirements at the E-MAD facility. The staffing requirements for dry storage would be about 120 full time employees. Considering that the total work force at the Nevada Test Site is approximately 4,000 (DOE, 1995g), the addition of 120 full time employees for foreign research reactor spent nuclear fuel storage is not expected to have any measurable socioeconomic impact in the region of influence.

F.4.5.1.2 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. For the purpose of these calculations, the refurbished E-MAD facility is treated as a generic dry storage facility. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6. Table F-84 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-84 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at:				
• E-MAD (dry storage)	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at:				
• E-MAD (dry storage)	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 5A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory or the Savannah River Site and 193 shipments directly from ports into a dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-85 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-85 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (person-rem)	*Worker Population Risk (LCF)*
	E-MAD	*E-MAD*
Phase 2	113	0.05

F.4.5.1.3 Material, Utility, and Energy Requirements

The material, utility, and energy requirements for the E-MAD facility are typical of those for dry storage. Table F-86 presents the estimated material, utility and energy consumption for dry storage.

Table F-86 Annual Utility and Energy Requirements for Dry Storage at the Nevada Test Site

Commodity	Baseline Site Usage	Dry Storage Usage	Percent Increase
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent 0.04 percent

[a] *The majority of the energy used at the Nevada Test Site is provided by electricity. Current usage is not available*

[b] *During receipt and handling*

[c] *During storage*

These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

F.4.5.1.4 Waste Management

The contribution of waste associated with the operation of the E-MAD facility is typical of that for new dry storage (Section F.4.5.2.1.13).

F.4.5.1.5 Air Quality

The contribution of air emissions associated with the operation of the E-MAD facility is typical to that for new dry storage (Section F.4.5.2.1.5).

F.4.5.1.6 Water Resources

The effect of the operation of the E-MAD facility on the water usage is typical to that for new dry storage (Section F.4.5.2.1.7).

F.4.5.2 New Facilities (Phase 2)

Analysis options 5A and 5B involve the use of new facilities as discussed above. The environmental impacts analyzed relate to the construction and operation of these facilities. The impacts include: land use; socioeconomics; cultural resources; aesthetic and scenic resources; geology; air and water quality; ecology; noise; traffic and transportation; occupational and public health and safety; materials, utilities, and energy; and waste management.

F.4.5.2.1 Dry Storage

Analysis option 5A involves long-term dry storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This analysis option would require the construction of a new dry storage facility. The analysis option encompasses both the dry vault design and the dry cask design as described in Section 2.6.5 and earlier in this appendix. There are no environmental impact parameters that would discriminate between the two designs.

F.4.5.2.1.1 Land Use

A new dry storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 3.7 ha (9 acres) of land. A new dry storage facility would occupy 5,000 m^2 (54,000 ft^2) of land and would move 11,000 m^3 (14,400 yd^3) of soil. Neither construction nor operation of a new dry storage facility at any of the areas would significantly impact land use patterns on the Nevada Test Site.

F.4.5.2.1.2 Socioeconomics

As discussed in Section F.3.1.1 the total capital cost of a new dry storage facility is estimated to be \$370 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$92.5 million. This represents approximately 66 percent of the estimated FY 1995 total expenditures for the Nevada Test Site (141 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be positive. The annual operations costs of a new dry storage facility are estimated to be \$15.6 million for receipt and handling and \$0.6 million for storage. These costs represent about 11 percent and 0.5 percent of FY 1995 total expenditures for the Nevada Test Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be small.

Direct employment associated with construction of a new dry storage facility is estimated to be 190 persons. The relative socioeconomic impact from direct construction employment on the region of influence would not be significant. In addition, when compared to the projected FY 1995 work force at the Nevada Test Site of approximately 4,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with receipt and storage operations is estimated to be 30 persons. Upon completion of these activities, direct employment is expected to decrease to eight persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Nevada Test Site.

F.4.5.2.1.3 Cultural Resources

There are no known historical, archaeological, paleontological, or Native American traditional sites in or around the potential storage site. No impacts to cultural resources are expected from ground disturbance, noise, or air emissions during facility construction or operation of the new dry storage facility. Consultation with the Nevada State Historic Preservation Office prior to project implementation is required under Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend further studies of the proposed storage site to verify that no archaeological areas would be disturbed by construction activities (DOE, 1995g).

F.4.5.2.1.4 Aesthetic and Scenic Resources

Construction and operation of a new dry storage facility for foreign research reactor spent nuclear fuel would have less impact on aesthetic and scenic resources at the Nevada Test Site than the construction of facilities for spent nuclear fuel management under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS (DOE, 1995g).

The proposed spent nuclear fuel facilities under Centralization, when fully constructed and under operation, would consist of a series of industrial buildings set within a 36-ha (90-acre) site. The site would not be visible from areas outside the Nevada Test Site. The new dry storage facility for foreign research reactor spent nuclear fuel would be constructed and operated under similar conditions. Potential soil erosion and dust generation associated with construction-related activities would be controlled by the implementation of best-management practices. Any visibility impacts from fugitive dust generation by construction-related activities should be insignificant and short term. Facility operations associated with the dry storage of foreign research reactor spent nuclear fuel should not generate any atmospheric emissions which would reduce area visibility.

F.4.5.2.1.5 Geology

The new dry storage facility for foreign research reactor spent nuclear fuel would be situated on tertiary volcanic or sedimentary rocks near volcanic or intrusive centers where small to medium-size precious metal deposits could be developed. However, because the Nevada Test Site is closed to mining operations, any precious metal deposits that might exist in or around the potential storage site would not be impacted (DOE, 1995g). Further, no mass movement or subsidence and sediment runoff from land disturbances would be expected (DOE, 1995g). The operation of the new dry storage facility would have no effect on the geologic characteristics at the site.

F.4.5.2.1.6 Air Quality

Nonradiological Emissions: Potential air quality impacts at the Nevada Test Site associated with the dry storage facility include the generation of fugitive dust from construction activities (e.g., clearing of land, grading, and road preparation) and vehicle emissions from the heavy equipment utilized during the construction phase of the project. Sources of fugitive dust include:

- transfer of soil to and from haul trucks and storage piles;
- turbulence created by construction vehicles moving over cleared, unpaved surfaces; and
- wind-induced erosion of exposed, barren surfaces.

The construction of this facility would require the clearing of 3.7 ha (7 acres) of land. However, the overall construction impacts to the ambient air quality of the region should be minimal due to the short duration (3 months to 6 years) of the project. Emissions of sulfur dioxide, nitrogen dioxide, and carbon monoxide are assumed to result entirely from diesel exhaust during the construction process. Respirable particulate matter (e.g., PM_{10}) is assumed to be 64 percent of the total suspended particulates estimated for the construction effort. Additionally, wetting controls are assumed to reduce this amount by 50 percent, which is a very conservative estimate.

Table F-87 presents the air quality impacts associated with the construction of the dry storage facility at the Nevada Test Site. Additionally, this table shows that the ambient impacts would be minimal and compliance with existing Federal and State ambient air quality standards would not be adversely affected. Therefore, construction activities would not be expected to have any detrimental effect on the health and safety of the general population. The estimated impacts from construction activities were generated using

Table F-87 Estimated Maximum Concentrations of Criteria Pollutants at the Nevada Test Site Attributable to New Dry Storage Construction

Pollutant	*Averaging Time*	*Ambient Standard*[b]	*Baseline Concentration*[a]	*Construction Activities*
Boundary (μg/m3):[b, c]				
• Particulate Matter (PM_{10})[a]	24-hour	150	84.90	0.0020
	Annual	50	0.43	0.1107
• Carbon Monoxide	1-hour	40,000	2,748.0	26.756
	8-hour	10,000	2,290.8	3.345
• Sulfur Dioxide	3-hour	1,300	170.3	2.356
	24-hour	365	55.2	0.345
	Annual	80	1.1	0.006
• Nitrogen Oxides	Annual	100	[d]	0.098

[a] *Source: (DOE, 1995g)*

[b] *64 percent of total suspended particulates is considered to be respirable particulate matter (e.g., PM_{10}) for the construction activities. The standard refers to the actual PM_{10} standard.*

[c] *To convert to μg/ft^3, multiply by 0.0283*

[d] *No sources indicated*

the U.S. Environmental Protection Agency's regulatory-approved Industrial Source Complex Short-Term Model, Version 2.0 in conjunction with onsite meteorological data from 1991.

Nonradiological emissions are not expected during operation of the new dry storage facility for foreign research reactor spent nuclear fuel. Any emissions associated with dry storage would be directly attributable to front-end wet storage activities only.

Radiological Emissions: No radiological emissions from construction of a new dry storage facility for foreign research reactor spent nuclear fuel are expected. Based on dry fuel drying and storage operations conducted at Idaho National Engineering Laboratory, potential atmospheric releases from the spent nuclear fuel storage facility would consist of minor amounts of particulate radioactive material and larger amounts of gaseous fission products that could escape from the fuel through cladding defects. The majority of radioactive material responsible for fuel and cask internal surface contamination consists of activation products that plate out on the spent nuclear fuel assemblies during reactor operation. This material is dependent on corrosion of structural materials and generally consists of radionuclides such as ^{58}Co, ^{60}Co, ^{50}Fe, etc. This contamination activity would have to be controlled during the cask opening and fuel handling operations to prevent internal personnel exposures. Proper facility ventilation (designed to provide airflow from areas of low contamination to progressively higher contamination) would help provide contamination control. High-efficiency particulate air filters in the facility exhaust would reduce the airborne effluent quantities of this particulate material to quantities that are well within the prescribed limits.

Cask opening and fuel drying operations may also be responsible for the release of significant amounts of ^{3}H, ^{85}Kr, and minor amounts of ^{129}I. The amount of these radionuclides that are released during the cask opening operation depends on the following parameters: (1) the number of spent nuclear fuel clad defects; (2) the spent nuclear fuel material and the diffusion rate of these radionuclides through the fuel matrix for the fuel temperature while in the cask; and (3) the time that the spent nuclear fuel is contained within the cask before opening.

Similarly, for fuel drying operations, the temperature of the drying gas (as well as the parameters discussed above) would cause quantities of ^{3}H, ^{85}Kr, and ^{129}I to be released from the fuel. Charcoal or silver zeolite filters could be used to remove the ^{129}I from the exhaust, but the ^{3}H and ^{85}Kr, being gases, or a gaseous state for the case of tritiated water, would be exhausted to the atmosphere. During spent nuclear fuel storage, small amounts of the gaseous/volatile radionuclides are expected to be released to the environment based on the fuel matrix, clad defects, and storage temperature. Release rates would decrease with storage time due to radioactive decay. It is anticipated that the fuel drying operation would be responsible for the most significant release of these gaseous/volatile radionuclides to the environment.

For this analysis, radiological emissions from the operation of a new dry storage facility for foreign research reactor spent nuclear fuel were calculated based on the methodology and assumptions described in Appendix F, Section F.5. The radiological consequences of air emissions from the operation of the new dry storage facility at the Nevada Test Site are discussed in Section F.4.5.2.1.11. The annual emission releases from the dry storage facility during receipt and unloading and storage are provided in Section F.6.6.1.

F.4.5.2.1.7 Water Resources

The water usage during construction of a new dry storage facility is estimated to be about 7.75 million l (2 million gal). During operations, annual water consumption would be 2.1 million l (550,000 gal) for receipt and handling and 0.4 million l (109,000 gal) for storage. With an annual average water usage of approximately 1,138 million l (301 million gal) for the Nevada Test Site, these amounts represent no more than a 0.2 percent increase in annual water usage. Therefore, a new dry storage facility would have minimal impacts on water resources at the Nevada Test Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Nevada Test Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Nevada Test Site could accommodate any new domestic and process wastewater streams from a new dry storage facility. The expected total flow volumes at the Nevada Test Site would still be well within the design capacities of treatment systems at the Nevada Test Site. A new dry storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.5.2.1.8 Ecology

Terrestrial Resources: Radiation doses received by terrestrial biota from foreign research reactor spent nuclear fuel activities would be expected to be similar to those received by man. Although guidelines have not been established for acceptance limits for radiation exposure to species other than man, it is generally agreed that the limits established for humans are also conservative for other species. Evidence indicates that no other living organisms have been identified that are likely to be substantially more radiosensitive than man. Thus, so long as exposure limits protective of man were not exceeded, no significant radiological impact on populations of biota would be expected as a result of spent nuclear fuel activities at Area 5 (DOE, 1995g).

Wetlands: Under the Centralization Alternative, construction of a new dry storage facility would result in the disturbance of approximately 36 ha (90 acres) [3.7 ha or (9 acres) for foreign research reactor spent nuclear fuel], or less than 1 percent of site location. No wetlands are expected to be disturbed because none exist in or around the proposed storage site (DOE, 1995g).

Threatened and Endangered Species: The project area is located within the range of the desert tortoise, a Federally-listed threatened species. Recent pre-activity surveys for other nearby projects have not identified the desert tortoise in the general area of the project site. However, a pre-activity survey for this project would be conducted to determine the presence or absence of the desert tortoise and other species of concern. If present, the desert tortoise could be impacted during construction of the new dry storage facility due to increased vehicular traffic, construction of trenches for utilities, and other temporary construction excavations. Prior to and during construction activities, fencing of the area and removal of tortoises within the fence would decrease the potential to bring harm to the desert tortoise (DOE, 1995g).

DOE has completed consultations with the U.S. Fish and Wildlife Service regarding threatened and endangered species for the proposed construction site of foreign research reactor spent nuclear fuel storage facilities at the Nevada Text Site, as required by the Endangered Species Act.

Construction of a new dry storage facility would have some adverse effects on animal populations. Less mobile animals, such as amphibians, reptiles, and small mammals within the project area would be destroyed during land-clearing activities. Larger mammals and birds in construction and adjacent areas would be disturbed by construction activities and would move to nearby suitable habitat. The long-term survival of these animals would depend on whether the area to which they moved was at or below its carrying capacity. Areas that would be revegetated upon completion of construction would be of minimal value to most wildlife but may be repopulated by more tolerant species (DOE, 1995g).

The Migratory Bird Treaty Act is primarily concerned with the destruction of migratory birds, as well as their eggs and nests. It may be necessary to survey construction sites for the nests of migratory birds prior to construction and/or avoid clearing operations during the breeding season (DOE, 1995g).

Activities associated with operation, such as noise, increased human presence and traffic, and night lighting could affect wildlife living immediately adjacent to the site. While these disturbances may cause some sensitive species to move from the area, most animals should be able to adjust.

F.4.5.2.1.9 Noise

Noises generated on the Nevada Test Site do not propagate offsite at levels that impact the general population. Thus, noise impacts for both the Centralization and Regionalization by Fuel Type and/or Geography Alternatives at the Nevada Test Site would be limited to those resulting from the transportation of personnel and materials to and from the site that affect nearby communities, and those resulting from onsite sources that may affect some wildlife near these sources (DOE, 1995g).

The transportation noises are a function of the size of the work force (e.g., an increased work force would result in increased employee traffic and corresponding increases in deliveries by construction crews). Such noise and activity associated with construction would be expected to have short-term effects on most wildlife (DOE, 1995g).

Under the Centralization Alternative, the projected Nevada Test Site work force would increase by about 48 percent in the years 2000 to 2002, during peak construction, and would decrease thereafter. There would be a corresponding increase in truck, private vehicle, and bus trips to the site. The day-night average sound level at 15 m (50 ft) from U. S. Route 95 would be expected to increase by about 1 decibel.

No change is expected in the community reaction to noise along this route. No mitigation of traffic noise impacts is proposed (DOE, 1995g).

F.4.5.2.1.10 Traffic and Transportation

Construction and operation of a new dry storage facility would involve a small increase in the number of employees commuting to the Nevada Test Site and transportation of foreign research reactor spent nuclear fuel and hazardous chemicals within the site.

The maximum reasonably foreseeable scenario for construction and operations traffic occurs under the Centralization Alternative considered in the Programmatic SNF&INEL Final EIS. This would occur in 2001, when there would be about 3,400 full-time employees, and about 1,200,000 people in the region of influence. None of the future baseline levels of service would change due to spent nuclear fuel-related impacts (DOE, 1995g). These conclusions are equally valid for a new dry storage facility for foreign research reactor spent nuclear fuel.

F.4.5.2.1.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an 80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.5 of this appendix. Table F-88 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-88 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site (New Dry Storage)

Facility	*MEI Dose (mrem/yr)*	*MEI Rise (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Dry Storage Facility	0.00076	3.8×10^{-10}	0.00093	4.7×10^{-7}
Storage at: • New Dry Storage Facility	0	0	0	0

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask). Analysis option 5A involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site, and 193 shipments directly from ports to a new dry storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-89 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative or regulatory limit at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-89 Handling-Related Impacts to Workers at the Nevada Test Site

	Worker Population Dose (Person-rem)		*Worker Population Risk (LCF)*	
	New Dry Storage Cask	*New Dry Storage Vault*	*New Dry Storage Cask*	*New Dry Storage Vault*
Phase 2	266	113	0.11	0.05

F.4.5.2.1.12 Material, Utility, and Energy Requirements

Construction of a new dry storage facility at the Nevada Test Site would consume 21,800 m^3 (28,500 yd^3) of concrete and 5,200 metric tons (5,750 tons) of steel. The total energy and water requirements during construction are estimated to be 835,000 l (221,000 gal) for fuel, and 7.75 million l (2 million gal) for water.

The annual utility and energy requirements during operations are shown in Table F-90. These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-90 Annual Utility and Energy Requirements for New Dry Storage at the Nevada Test Site

Commodity	*Baseline Site Usage*	*Dry Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.6 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,138,000,000	1,590,000[b] 400,000[c]	0.14 percent[b] 0.04 percent[b]

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.1.13 Waste Management

Construction of a new dry storage facility at the Nevada Test Site would generate 1,800 m^3 (2,400 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-91. These quantities represent a very small percent increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new dry storage facility. Therefore, the impact of this waste on existing Nevada Test Site waste management capacities would be minimal.

Table F-91 Annual Waste Generated for New Dry Storage at the Nevada Test Site

Waste Form	*Baseline Site Generation*	*Dry Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	22[a] 1[b]	0.20 percent[a] 0.01 percent[b]
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.4 percent[a] 3.6 percent[b]

[a] *During receipt and handling*

[b] *During storage*

F.4.5.2.2 Wet Storage

Analysis option 5B involves long-term wet storage of foreign research reactor spent nuclear fuel at the Nevada Test Site. This storage option would require the construction of a new wet storage facility.

F.4.5.2.2.1 Land Use

A new wet storage facility would be located in Area 5 in the southeastern portion of the Nevada Test Site. The land in this area can be characterized as sparsely vegetated desert, ready for development. Use of Area 5 for foreign research reactor spent nuclear fuel storage would be consistent with existing land use plans, which designate this area for general use. Construction activities, including laydown areas, would disturb 2.8 ha (7 acres) of land. A new wet storage facility would occupy 3,800 m^2 (41,000 ft^2) of land and would move 18,000 m^3 (24,000 yd^3) of soil. Neither construction nor operation of a new wet storage facility at any of the areas would significantly impact land use patterns on the Nevada Test Site.

F.4.5.2.2.2 Socioeconomics

As discussed in Section F.3.2 the total capital cost of a new wet storage facility is estimated to be \$449 million. Construction activities are projected to take 4 years. Assuming that the capital cost is evenly distributed over this 4-year period, the annual expenditures would be about \$112.2 million. This represents approximately 79.5 percent of the estimated FY 1995 total expenditures for the Nevada Test Site (141 million). The relative socioeconomic impact from annual construction expenditures on the region of influence would be substantial. The annual operations costs of a new wet storage facility are estimated to be \$23.3 million for receipt and handling and \$3.5 million for storage. These costs represent about 16.5 percent and 2.5 percent of FY 1995 total expenditures for the Nevada Test Site. The relative socioeconomic impact from annual operation expenditures on the region of influence would be positive.

Direct employment associated with construction of a new wet storage facility is estimated to be 157 persons. The relative socioeconomic impact from direct construction employment on the region of influence would be small. In addition, when compared to the projected FY 1995 work force at the Nevada Test Site of approximately 4,000 persons, the relative socioeconomic impact of this temporary increase in construction employment would be insignificant. Direct employment associated with operations of a new wet storage facility is estimated to be 30 persons. The relative socioeconomic impact of this increase in operations employment would be insignificant to both the region of influence and the Nevada Test Site.

F.4.5.2.2.3 Cultural Resources

Impacts to cultural resources would be the same as for new dry storage (Section F.4.5.2.1.3).

F.4.3.2.2.4 Aesthetic and Scenic Resources

Impacts to aesthetic and scenic resources would be the same as for new dry storage (Section F.4.5.2.1.4).

F.4.5.2.2.5 Geology

Impacts to geology would be the same as for new dry storage (Section F.4.5.2.1.5).

F.4.5.2.2.6 Air Quality

Nonradiological Emissions: Construction of a new wet storage facility would necessitate the clearing and grading of approximately 3 ha (7 acres) of land. In comparison, approximately 4 ha (10 acres) of land would be disturbed by new dry storage construction. Therefore, air quality impacts associated with wet storage construction would be bound by those associated with dry storage construction, as presented in Section F.4.5.2.1.6.

No nonradiological emissions from the operation of the new wet storage facility are expected.

Radiological Emissions: Incident-free airborne releases from the new wet storage facility would be limited to radioactive noble gases and some radioactive iodine which could be released from the stored fuel prior to canning. The airborne materials released to the building atmosphere during incident-free operations would be filtered by the building heating and ventilation system. Radioactive and nonradioactive effluent gases would be routed through double-banked high-efficiency particulate air filters prior to release to the environment through an exhaust air system. The high-efficiency particulate air filter would have a minimum efficiency of 99.97 percent for 0.3 micron diameter particulates and would allow in-place dioctyl phthalate testing.

The new wet storage facility would discharge all ventilated gas, except truck exhaust, to the facility exhaust system. Truck exhaust would be discharged directly to the environment during cask off-loading operations in the truck receiving area. The exhaust air system would employ a detector to monitor ^{137}Cs. For other building areas which would be sources of airborne radioactive contamination, the heating, ventilating, and air conditioning system would be designed to maintain airflow from areas of low potential contamination into areas of higher potential contamination. These airborne effluents would be required to be below the radioactivity concentration guides listed in DOE Order 5480.1B for both onsite and offsite concentrations (DOE, 1989b).

Air emissions from the new wet storage facility are expected to be similar to the air emissions from the CPP-603 at Idaho National Engineering Laboratory. The annual air emission for the CPP-603 was designed to result in ground-level concentrations of less than 0.003 percent of DOE 5480.1B limits for uncontrolled areas. Radiological emissions from the operation of the wet storage facility were calculated based on the methodology and assumptions used in Appendix F, Section F.6. The annual emission releases from the wet storage facility during the receipt and unloading, and storage are provided in Section F.6.6.1.

F.4.5.2.2.7 Water Resources

The annual water usage during construction and operation of a new wet storage facility is estimated to be about 1.9 million l (1.2 million gal) and 2.7 million l (720,000 gal), respectively. With an annual average water usage of approximately 1,138 million l (301 million gal) for the Nevada Test Site, these amounts represent an increase of about 0.17 percent and 0.23 percent, respectively. Therefore, a new wet storage facility would have minimal impact on water resources at the Nevada Test Site.

Best-management practices during construction would prevent sediment runoff or spills of fuels or chemicals. Therefore, construction activities should have no impact on water quality at the Nevada Test Site. The impact on water quality during operations would also be negligible. Existing water treatment facilities at the Nevada Test Site could accommodate any new domestic and process wastewater streams from a new wet storage facility. The expected total flow volumes at the Nevada Test Site would still be well within the design capacities of treatment systems at the Nevada Test Site. A new wet storage facility would meet National Pollutant Discharge Elimination System limits and reporting requirements, so no impact on the water quality of receiving streams is expected.

F.4.5.2.2.8 Ecology

Impacts to the ecology would be the same as for new dry storage (Section F.4.5.2.1.8).

F.4.5.2.2.9 Noise

Impacts from noise would be the same as for new dry storage (Section F.4.5.2.1.9).

F.4.5.2.2.10 Traffic and Transportation

Impacts from traffic and transportation would be the same as for new dry storage (Section F.4.5.2.1.10).

F.4.5.2.2.11 Occupational and Public Health and Safety

Emission-Related Impacts: Doses that could be received by the public during incident-free operation associated with the receipt and management of the foreign research reactor spent nuclear fuel at the Nevada Test Site would be attributed to emissions of radioactive material that could be carried by wind offsite. The general public would be too far from the locations where handling activities or storage take place to receive any dose from direct exposure. Doses were calculated for the MEI, defined as an individual at the site boundary receiving the maximum exposure, and for the general population within an

80 km (50 mi) radius of the storage facility. These doses would result from incident-free airborne radiological emissions assumed to be released from the unloading of the transportation cask and the storage facility during storage. The methodology and assumptions used for the calculation of the radiological emissions and resulting doses are discussed in Section F.6 of this appendix. Table F-92 summarizes the annual emission-related doses to the public and the associated risks for the MEI and population at the Nevada Test Site. Integrated doses for the duration of a specific implementation period can be obtained by multiplying the annual dose by the number of years in the period.

Table F-92 Annual Public Impacts for Foreign Research Reactor Spent Nuclear Fuel Receipt and Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Facility	*MEI Dose (mrem/yr)*	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Receipt/Unloading at: • New Wet Storage Facility	0.00052	2.6×10^{-10}	0.00052	2.6×10^{-7}
Storage at: • New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.4×10^{-12}

Handling-Related Impacts: Workers at the site would receive radiation doses during handling operations (i.e., receiving and unloading the transportation cask), transferring the foreign research reactor spent nuclear fuel from one facility to another, or preparing the foreign research reactor spent nuclear fuel for shipment offsite. Analysis option 5B involves the receipt of 161 shipments of foreign research reactor spent nuclear fuel from the Idaho National Engineering Laboratory and/or the Savannah River Site and 193 shipments directly from ports into a new wet storage facility. The assumptions and methodologies used to calculate the doses to a working crew associated with the handling activities of the foreign research reactor spent nuclear fuel are described in Section F.5 of this appendix.

Table F-93 presents the population dose and risk that would be received by the members of the working crew if that working crew handled the total number of transportation casks at the Nevada Test Site. The worker MEI doses and risks were not calculated because of the large uncertainties associated with the assumptions for such calculations. However, the upper bound for such a dose would be equal to the administrative limits at the site. For DOE radiation workers, the regulatory limit is 5,000 mrem per year. All these workers would be monitored and if any worker's dose approached this limit, he or she would be rotated into a different job to prevent further exposure. This regulatory limit provides a very conservative upper bound on the radiation dose for the worker MEI. If a single worker received the full 5,000 mrem per year dose for the full 13 years of potential foreign research reactor spent nuclear fuel receipt, then the MEI dose would be 65,000 mrem. For this dose, the associated risk of incurring an LCF would be 2.6 percent.

Table F-93 Handling-Related Impacts to Workers at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Worker Population Dose (Person-rem)	*Worker Population Risk (LCF)*
	New Wet Storage	*New Wet Storage*
Phase 2	109	0.04

F.4.5.2.2.12 Material, Utility, and Energy Requirements

Construction of a new wet storage facility at the Nevada Test Site would consume 12,400 m^3 (16,260 yd^3) of concrete and 3,100 metric tons (3,443 tons) of steel. The total energy and water requirements during construction are estimated to be 600,000 l (159,000 gal) for fuel, and 4.4 million l (1.2 million gal) for water. The annual utility and energy requirements during operations are shown in Table F-94. These requirements represent a small percent of current requirements for the Nevada Test Site. No new generation or treatment facilities would be necessary, and connections to existing networks would require only short tie-in lines. Increases in consumption would be minimal because overall activity on the Nevada Test Site is expected to decrease because of changes in site mission and a general reduction in employment.

Table F-94 Annual Utility and Energy Requirements for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Commodity	*Baseline Site Usage*	*Wet Storage Usage*	*Percent Increase*
Electricity (MW-hr/yr)	176,440	800 - 1,000	0.84 percent
Fuel (l/yr)	[a]	0	0 percent
Water (l/yr)	1,139,000,000	2,700,000[b] 1,500,000[c]	0.23 percent 0.13 percent

[a] *The majority of energy used at the Nevada Test Site is provided by electricity.*

[b] *During receipt and handling*

[c] *During storage*

F.4.5.2.2.13 Waste Management

Construction of a new wet storage facility at the Nevada Test Site would generate 2,600 m^3 (10,300 yd^3) of debris. The annual quantities of waste generated during operations are shown in Table F-95. These quantities represent a very small percentage increase above current levels at the Nevada Test Site. Existing waste management storage and disposal activities at the Nevada Test Site could accommodate the waste generated by a new wet storage facility. Therefore, the impact of this waste on the existing the Nevada Test Site waste management capacities would be minimal.

Table F-95 Annual Waste Generated for Wet Storage at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

Waste Form	*Baseline Site Generation*	*Wet Storage Generation*	*Percent Increase*
High-Level (m^3/yr)	0	none	0 percent
Transuranic (m^3/yr)	0	none	0 percent
Solid Low-Level (m^3/yr)	10,845	16[a] 1[b]	0.15 percent 0.01 percent
Wastewater (l/yr)	11,000,000	1,590,000[a] 400,000[b]	14.5 percent 3.6 percent

[a] *During receipt and handling*

[b] *During storage*

F.4.5.3 Accident Analysis

An evaluation of incident-free operations and hypothetical accidents at the Nevada Test Site is presented here based on the methodology in Appendix F, Section F.6. The evaluation assessed the possible radiation exposure to individuals and general population due to the release of radioactive materials. The analyses are based on the same operations carried out at the different potential storage locations and the same accidents at any of the sites evaluated. Information concerning radiological doses to individuals and the general population are the same as set forth in Section F.4.1.3.

Table F-96 presents the frequency and consequences in terms of mrem or person-rem, of postulated accidents to the offsite MEI, NPAI, and offsite population for the 95th-percentile meteorological conditions using the assumptions and input values discussed above. The worker doses are calculated only for the 50th-percentile meteorology. This is an individual assumed to be 100 m (330 ft) downwind of the accident. DOE did not estimate the worker population dose.

Table F-96 Frequency and Consequences of Accidents at the Nevada Test Site

	Frequency (per year)	*Consequences*			
		MEI (mrem)	*NPAI (mrem)*	*Population (person-rem*	*Worker (mrem)*
Dry Storage Accidents[a]					
• Spent Nuclear Fuel Assembly Breach	0.16	1.7	0.31	1.5	20
• Dropped Fuel Cask	0.0001	0.11	0.0014	0.40	0.089
• Aircraft Crash w/Fire	1×10^{-6}	180	1.2	250	87

[a] *E-MAD and New Dry Storage Facility*

Multiplying the frequency of each accident times its consequences and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Nevada Test Site. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks for the Nevada Test Site. These risk estimates are presented in Table F-97.

Table F-97 Annual Risks of Accidents at the Nevada Test Site

	Consequences			
	MEI (LCF/yr)	*NPAI (LCF/yr)*	*Population (LCF/yr)*	*Worker (LCF/yr)*
Dry Storage Accidents[a]				
• Spent Nuclear Fuel Assembly Breach	1.4×10^{-7}	2.5×10^{-8}	0.00012	0.0000013
• Dropped Fuel Cask	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	3.6×10^{-12}
• Aircraft Crash w/Fire	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	3.5×10^{-11}

[a] *E-MAD and New Dry Storage Facility*

Table F-98 presents the frequency and consequences of the accidents analyzed for each site for wet storage (Implementation Alternative 5 of Management Alternative 1). Multiplying the frequency of each accident times its consequences at each site and converting the radiation doses to LCF yields the annual risks associated with each potential accident at the Nevada Test Site. These annual risks are multiplied by the maximum duration of this implementation alternative at each site to obtain conservative estimates of risks at the Nevada Test Site. Table F-99 presents the risk estimates from this implementation.

F.4.5.3.1 Secondary Impact of Radiological Accidents at the Nevada Test Site

In the event of an accidental release of radioactivity, there is a potential for impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies (secondary impacts). For this analysis, secondary impacts of radiological accidents involving foreign research reactor spent nuclear fuel have been qualitatively assessed based on the calculations presented in Section F.4.5.3. Radiological

Table F-98 Frequency and Consequences of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Frequency (per year)	Consequences			
		MEI (mrem)	NPAI (mrem)	Population (person-rem)	Worker (mrem)
• Spent Nuclear Fuel Assembly Breach	0.16	0.054	0.0016	0.33	0.10
• Accidental Criticality	0.0031	88	15	54	1,300
• Aircraft Crash	1×10^{-6}	29	4.2	61	290

Table F-99 Annual Risks of Accidents at the Nevada Test Site (Implementation Alternative 5 of Management Alternative 1)

	Consequences			
	MEI (LCF/yr)	NPAI (LCF/yr)	Population (LCF/yr)	Worker (LCF/yr)
• Spent Nuclear Fuel Assembly Breach	4.2×10^{-9}	1.3×10^{-10}	0.000026	6.4×10^{-9}
• Accidental Criticality	1.4×10^{-7}	2.3×10^{-8}	0.000084	0.000016
• Aircraft Crash	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	1.2×10^{-10}

accidents that resulted in doses to the MEI of less than the annual Federal radiological exposure limit for the public of 100 mrem (10 CFR Part 20) were considered to have no secondary impacts.

The MEI dose provides a measure of the air concentration and radionuclide deposition at the receptor location. As such, it can be used to express the level of contamination from a given radiological accident. In estimating the human health effects from radiological exposure (as presented in Section F.4.1.3), the MEI dose evaluates four pathways: (1) air immersion, (2) ground surface, (3) inhalation, and (4) ingestion. In estimating the environmental effects from radiological exposure, however, only the air immersion and ground surface pathways need be considered.

At the Nevada Test Site, the radiological accident with the highest MEI dose is the aircraft crash into a dry storage facility with fire (Table F-96). For this accident, the MEI dose would be 180 mrem. For the air immersion and ground surface pathways only, the dose would be 1.0 mrem, which is less than the 100 mrem limit used in this analysis. Therefore, no secondary impacts to land uses, cultural resources, water quality, ecology, national defense, and local economies from radiological accidents involving foreign research reactor spent nuclear fuel storage would be expected at the Nevada Test Site.

F.4.5.4 Cumulative Impacts at the Nevada Test Site

The section presents the cumulative impacts of the proposed action, potential impacts of other contemplated DOE actions, and current activities at the site. A major portion of the presentation is based on information included in the Programmatic SNF&INEL Final EIS (DOE, 1995g) and the Tritium Supply and Recycling Final EIS (DOE, 1995a). The Programmatic SNF&INEL Final EIS includes the quantitative impacts from a proposed Expended Core Facility at the site. The Nevada Test Site is also considered in the storage and disposition of weapons-usable fissile materials program which could affect the site environment. The impacts from this program have not been determined sufficiently at this time to allow impact evaluation.

Table F-100 Cumulative Impacts at the Nevada Test Site

Environmental Impact Parameter	*FRR SNF Contribution*	*Other Activities*[a]	*Cumulative Impact*
Land Use (acres)	9	314,393[b]	314,402
Socioeconomics (persons)	190[c]/30[d]	2,662[c]/1,000[d]	2,852[c]/1,030[d]
Air Quality (nonradiological)	See Table F-100A	See Table F-100A	See Table F-100A
Occupational and Public Health and Safety			
• MEI Dose (rem/yr)	7.6×10^{-7}	0.00031	0.00031
LCF (per year)	3.8×10^{-10}	1.55×10^{-7}	1.55×10^{-7}
• Population Dose (person-rem/yr)	0.00093	0.095	0.095
LCF (per year)	4.7×10^{-7}	.00047	0.000047
• Worker Collective Dose (person-rem/yr)	8.9[e]	81	89.9
LCF (per year)	0.0036	0.032	0.035
Energy and Water Consumption			
• Electricity (MW-hr/yr)	1,000	4,019,000[f]	4,020,000
• Fuel (million l/yr)	0	6,129	6,129
• Water (million l/yr)	2.2	2,563	2,565
Waste Generation			
• High-Level (m^3/yr)	0	0	0
• Low-Level (m^3/yr)	22	44,578	44,600
• Mixed/Hazardous (m^3/yr)	0	252	252
• Transuranic (m^3/yr)	0	16	16

[a] *Other activities include existing activities, DOE-owned spent fuel management activities, construction and operation of an Expended Core Facility, and construction and operation of a tritium production facility.*

[b] *Existing developed land area is 314,000 acres*

[c] *Increase over baseline (3,300 persons) during construction activities*

[d] *Increase over baseline (3,300 persons) during operation activities*

[e] *The dose is due to the handling of foreign research reactor spent nuclear fuel during receipt, averaged over 30 years*

[f] *Major portion is the requirement for electricity by the tritium production (accelerator) facility (3,740,000 MW-hr/yr)*

Tables F-100 and F-100A summarize the cumulative impacts for land use, socioeconomics, air quality, occupational and public health and safety, energy and water consumption and waste generation at the site. Table F-100 also presents the contribution from the storage of foreign research reactor spent nuclear fuel to the cumulative impacts at the Nevada Test Site. For the purposes of this analysis, both the contributions from management of foreign research reactor spent nuclear fuel and the cumulative impacts were maximized by selecting the Centralization Alternative of the Programmatic SNF&INEL Final EIS at the Nevada Test Site.

As shown in Table F-100, the contribution from storage of foreign research reactor spent nuclear fuel to the cumulative impacts (under the Centralization Alternative) at the Nevada Test Site would be minimal. The Programmatic SNF&INEL Final EIS concludes that the implementation of any of the alternatives (including the Centralization Alternative) for the DOE spent nuclear fuel management program would not be expected to significantly contribute to cumulative impacts (DOE, 1995g). This conclusion is also valid for the implementation of any of the alternatives considered in this EIS for storage of foreign research reactor spent nuclear fuel at the Nevada Test Site.

Table F-100A Estimated Maximum Nonradiological Cumulative Ground-Level Concentrations of Criteria and Toxic Pollutants at the Nevada Test Site[a]

Pollutant	*Averaging Time*	*Regulatory Standard (μg/m³)*	*Cumulative Concentration[b] (μg/m³)*
Carbon Monoxide	1-hour	40,000	2,815 (7%)
	8-hour	10,000	2,306 (23%)
Nitrogen Oxides	Annual	100	4.2 (4.2%)
Sulfur Dioxide	3-hour	1300	173.6 (13.3%)
	24-hour	365	55.5 (15.2%)
	Annual	80	1.1 (1.3%)
Particulate Matter (PM_{10})	24-hr	150	85 (56.6%)
	Annual	50	0.54 (1.1%)

[a] *Concentrations represent activities from: foreign research reactor spent nuclear fuel management, DOE-owned spent nuclear fuel management, construction and operation of an Expended Core Facility, and construction and operation of a tritium production and recycling facilities*

[b] *Number in parentheses indicate the percentage of the Regulatory Standard*

F.4.5.5 Unavoidable Adverse Environmental Impacts

Construction of the potential new foreign research reactor spent nuclear fuel storage facilities would require the disturbance of approximately 4 ha (10 acres) of undeveloped land. Although this represents less than one percent of the undeveloped land on the Nevada Test Site, it would eliminate potential terrestrial wildlife habitat, including habitat potentially suitable for the Federally-listed desert tortoise. It would also require the dedication of a small land parcel potentially suitable for other construction projects, but similar land parcels are abundant on the Nevada Test Site.

F.4.5.6 Irreversible and Irretrievable Commitments of Resources

Construction and operation of new foreign research reactor spent nuclear fuel facilities would require commitments of electrical energy, fuel, concrete, steel, sand, gravel and miscellaneous chemicals. Groundwater to operate the foreign research reactor spent nuclear fuel facilities would be withdrawn from an aquifer that is presently experiencing localized overdraught. Further studies would be necessary to quantify any irreversible effects on future groundwater availability attributable to spent nuclear fuel withdrawals from that aquifer. The land dedicated to the foreign research reactor spent nuclear fuel facilities would become available for other rural uses following closure and decommissioning.

F.4.5.7 Mitigation Measures

Mitigation is addressed in general terms and describes typical measures that the Nevada Test Site could implement. The analyses indicate that the environmental consequences attributable to foreign research reactor spent nuclear fuel management activities at the site would be minimal in most environmental media.

Pollution Prevention: The DOE Nevada Field Office published a Waste Minimization and Pollution Prevention Awareness Plan in June 1991 to reduce the quantity and toxicity of hazardous, mixed, and radioactive wastes generated at DOE Nevada Field Office facilities. The plan is designed to reduce the possible pollutant releases to the environment and thus increase the protection of employees and the public. All DOE Nevada Field Office contractors and the Nevada Test Site users that exceed the Environmental Protection Agency criteria for small-quantity generators are establishing their own waste minimization and pollution prevention awareness programs that are implemented by the DOE Nevada Field Office plan. Contractor programs ensure that waste minimization activities are in accordance with Federal, State, and local environmental laws and regulations, and DOE orders (DOE, 1995g).

Additional goals include the promotion and use of nonhazardous materials, establishment of a baseline of waste generation data, calculations of annual reductions of waste generated, and implementation of regulatory programs. Goals also include incorporation of waste minimization concepts and technologies in planning and design of new processes and facilities and in upgrades of existing facilities. A waste minimization task force composed of representatives from each contractor and the Nevada Test Site user has been established to coordinate DOE/Nevada Test Site waste minimization and pollution awareness activities (DOE, 1995g).

Socioeconomics: To reduce construction- and operation-related impacts, possible coordination with local communities could address potential impacts from increased labor and capital requirements. The knowledge of the extent and effect of growth due to foreign research reactor spent nuclear fuel management activities could greatly enhance the ability of affected jurisdictions to plan effectively. Effective planning would address changes in levels of service for housing, infrastructure, utilities, transportation, and public services and finances (DOE, 1995g).

To alleviate potential impacts associated with the in-migration of labor, local labor force availability could be increased through various employment training and referral systems currently provided by the Nevada Test Site. The goal of these systems would be to reduce the potential for in-migration of labor to support foreign research reactor spent nuclear fuel management activities (DOE, 1995g).

Cultural Resources: Consultation with the Nevada State Historic Preservation Office prior to project implementation is required under Section 106 of the National Historic Preservation Act of 1966. The State Historic Preservation Office may recommend that further archaeological studies be conducted throughout the construction area to verify that there are no archaeological sites subject to disturbance (DOE, 1995g).

Water Resources: The foreign research reactor spent nuclear fuel facilities would have to be located and constructed to minimize floodplain impacts and to avoid floodplains to the maximum extent possible, as required by Executive Order 11988 (Floodplain Management) and DOE orders. Site-specific surveys would be performed to determine locations of flooding elevations more accurately (DOE, 1995g).

Accidents: The foreign research reactor spent nuclear fuel storage facilities would be designed to comply with current Federal, State, and local laws, DOE orders, and industrial codes and standards. This would provide facilities that are highly resistant to the effects of severe natural phenomena, including earthquakes, floods, tornadoes, and high winds, as well as credible events appropriate to the site, such as fires and explosions and manmade threats to its continuing structural integrity for containing materials (DOE, 1995g).

An emergency preparedness plan would also be developed to lower the potential consequences of an accident to workers and the public. All workers receive evacuation training to ensure timely and orderly personnel movement away from high-risk areas. Plans and arrangements with local authorities would also be in place to evacuate the general public that may be at risk of exposure to hazardous materials accidentally released (DOE, 1995g).

F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Occupational exposure to gamma radiation would depend largely on the operational history of the spent nuclear fuel elements to be stored in the facility and the length of time that these elements have been allowed to decay from the time that they were taken out of the reactor until they were placed in the cask for shipment to the storage facility. Normally, the decay time for fuel elements is established so that the gamma heating in the transportation cask is within specification and the radiation field on the outside cask surface is 200 mrem per hour or less. Special shipments can be made, however, with higher cask surface radiation fields, provided other requirements are met. Radiation exposures to personnel during receiving operations and surveys would depend on the level of radiation that is measurable on the exterior (surface) of the transportation cask. These initial operations are anticipated to provide the majority of personnel exposure since the remaining operations would be remote and could take advantage of the shielding built into the facility.

Realistic annual occupational radiation exposure estimates for facility operation can be performed once the following have been established:

- determination of accurate decay-time averaged values for the spent nuclear fuel,
- development of shielding characteristics for transportation casks for the spent nuclear fuel to be shipped to the facility,
- definition of personnel requirements for each of the individual operations to be accomplished within the facility, and
- completion of a time-motion study for the spent nuclear fuel element movement through the preliminary design of the facility.

The analyses in this appendix are based on a best estimate of the above conditions. The potential impacts are given in doses per cask shipment, so that the results can be simply multiplied by the total number of shipments for each potential storage arrangement.

Wet Storage: Occupational radiation exposure from the receipt, handling and storage of foreign research reactor spent nuclear fuel at a wet storage facility is treated in a generic way for all potential management sites, since the activities are essentially identical regardless of where the facility is located. It is based on actual handling experience of spent nuclear fuel at the Idaho National Engineering Laboratory and the Savannah River Site.

The workers involved with each cask were assumed to include the shipping agent, shift foremen, health physics technicians, and equipment operators. The equipment operators include onsite workers who remove each cask from its shipping container and transport it to the receiving bay, and those who perform most of the actual labor involved thereafter, such as transferring the spent nuclear fuel to storage and decontaminating the empty cask prior to returning it to the owners. Thus, while the assessment does not distinguish between them, the operators are a diverse group of workers whose distinct duties make it unlikely that the same operators could receive all of the calculated individual doses discussed below. As a result, it was assumed that the two foremen and two operators involved in handling the spent nuclear fuel casks outside the receiving facility would be different than the two foremen and two operators working inside the facility (the health physics technicians were assumed to be the same). This provides a conservative estimate of 12 workers.

In order to estimate the occupational radiation doses from the handling of foreign research reactor spent nuclear fuel transportation casks at the spent nuclear fuel management sites, it was necessary to develop a curve of dose rate versus distance for these casks. Historical data based on 44 research reactor spent nuclear fuel transportation cask receipts at either the Savannah River Site or the Idaho National Engineering Laboratory were obtained and evaluated. This historical data showed an average measured dose rate of approximately 2.3 mrem per hour at 1 m (3.3 ft) from the surface of the transportation cask. One cask, however, was measured to be 20 mrem per hour at 1 m (3.3 ft). To encompass this historical data, including the highest measured dose rate cask, an analysis was performed that assumed a dose rate of 23 mrem per hour at 1 m (3.3 ft) from the cask surface. It should be noted that, in the unlikely event that a higher dose rate transportation cask was received at the management site, radiological control procedures for as low as reasonably achievable limits would be utilized to ensure that the worker doses would be minimized. Dose rate reduction is usually accomplished by a combination of restrictions on time, distance from the source, and the provision of additional radiation shielding.

The plot of bounding transportation cask dose rate versus distance in Figure F-50 was developed using the ZYLIND computer code and appropriate conservative methodology. ZYLIND (RSIC, 1990) is a shielding computer code that uses the point kernel method to calculate photon dose rates from a cylindrical source and shield geometry. ZYLIND was developed in Germany in 1989 and then released to the Oak Ridge National Laboratory Radiation Shielding Information Center. ZYLIND has been extensively validated by comparison to measured dose rates from several hundred cylindrical containers with radioactive materials. ZYLIND calculated dose rates that were conservative and within 10 to 20 percent of the measured dose rates. ZYLIND allows the photon energy source to be divided into up to 20 energy groups from 0 to 10 million electron volts (Mev), allows up to eight materials regions, and includes mass attenuation and dose buildup information for a wide range of shielding materials.

The methodology used in calculating bounding transportation cask dose rates had four underlying assumptions. First, it was assumed that the dose rate at 1 m (3.3 ft) from the cask surface is 23 mrem per hour. Second, neutron dose rates from foreign research reactor spent nuclear fuel were assumed to be negligible and the only dose was assumed to be due to gamma (photon) radiation. A third assumption was that the foreign research reactor spent nuclear fuel source term inside a cask could be conservatively simulated by a single 1.0 Mev gamma energy group. Traditional NRC source terms (DiNunno et al, 1962) for spent nuclear fuel fission products assume an average gamma energy of 0.7 Mev. By using 1.0 Mev, the average gamma energy is expected to be conservatively bounded. Finally, it was assumed that the use of a point kernel cylindrical source-shield computer code (i.e., ZYLIND) would conservatively calculate the dose rates from a transportation cask.

The principal inputs for the calculation were the ZYLIND computer code manual (Radiation Shielding Information Center ZYLIND) and the U.S. Department of Transportation Certificates of Competent Authority for seven transportation cask designs that are likely to be used for the shipment of foreign research reactor spent nuclear fuel. These seven designs are: TN7, GNS-11, LHRL-120, NAC/LWT, PEGASE (IU-04), BMI-1, and GE-2000. These transportation casks are described in Appendix B, Section B.2. The U.S. Department of Transportation Certificates of Competent Authority provided geometry data on the cask inside cavity dimensions and the thickness and material composition of shielding adjacent to the cavity for each design.

With the cask geometry information, a set of ZYLIND calculations was performed for each design. An initial 1.0 Mev gamma source was estimated and ZYLIND was executed to calculate the dose rate at 1 m (3.3 ft) from the cask surface. This source was iterated upon until the 1 m (3.3 ft) dose rate equaled 23 mrem per hour. After this source was determined, the same source and cask geometry were rerun to calculate the dose rate at distances of from 0-50 m (0-164 ft) from the cask surface. This process was repeated for each of the transportation cask designs. The resulting dose rates at distance for each cask design were compared and the highest dose rate response at all distances was synthesized from this data to produce Figure F-50.

Table F-101 shows the actual dose rates encountered during receipt and handling for essentially all of the foreign research reactor spent nuclear fuel casks, which are expected to be one to two orders of magnitude lower than the limit, based on actual experience with foreign research reactor spent nuclear fuel in the past.

Table F-102 presents the wet storage collective dose for unloading one transportation cask using time, distance, and personnel data from the Idaho National Engineering Laboratory and the dose rate curve in Figure F-50. The total worker dose per transportation cask was calculated to be 0.31 person-rem. The actual distances for each worker are based on conservative estimates of actual work experience that would reflect an as low as reasonably achievable Radiation Protection Program as required by DOE regulation (10 CFR 835).

Generic Dry Cask Storage: The receipt, handling, and storage occupational radiation doses (deep dose equivalents) for dry storage are also treated in a generic way, since the operation of the general facility designed for dry storage would be the same at any management site. The assessment is based on Pressurized Water Reactor spent nuclear fuel from the reactor's spent nuclear fuel storage pool to an NRC-licensed dry storage facility at the Calvert Cliffs nuclear power plant in Maryland. The system employed is the horizontal module system (NUHOMS), which was selected for this assessment for two reasons: (1) it is a current, regulatory-approved design that is readily available for foreign research reactor spent nuclear fuel dry storage, and (2) the worker dose rates calculated for the system are among the highest of the current systems now in use for storage of commercial spent nuclear fuel. As a result, the system analysis provides a reasonably conservative estimate for storage of foreign research reactor spent nuclear fuel in NUHOMS and a reasonable upperbound assessment for all other foreign research reactor spent nuclear fuel generic dry storage. The Calvert Cliffs Safety Analysis Report does not identify each category of worker associated with receipt, handling, transfer, and storage of spent nuclear fuel. As a result, for this assessment, the doses (deep dose equivalent) were assumed to be the same for all workers (titled "operators" in the assessment). However, it would appear from the work activities that it cannot be the same operators who support each of the activities. As a result, job titles comparable to those considered for wet storage have been defined in order to determine the average worker dose per cask. Therefore, it is assumed that the following categories of workers are involved: foremen (2), health physics technicians (2), equipment operators for the storage pool activities (2), and different equipment operators for the one-site transport and transfer of the spent nuclear fuel from the transfer cask to the dry storage cask (3), welders (2), helium leak test technician (1), and dye penetrant test technician (1). Each of the distances listed is the average distance for all of the workers involved in each one of the 25 specific activities associated with receipt, handling, transfer, and dry storage. Thus, for example, the first activity [loading fuel into the container (dry shielded canister)], would involve four workers (one foreman, one health physics technician, and two operators) in the Spent Fuel Pool area. The results indicate that the collective dose to the working crew of 13 would be 1.5 person-rem per NUHOMS cask transfer. A transfer cask load is approximately equal to the foreign research reactor spent nuclear fuel inventory of eight transportation casks.

IFSF (Dry Vault) Specific Dry Storage: Based on data provided by the Idaho National Engineering Laboratory, Table F-103 was generated to present the occupational dose for unloading one transportation cask into the IFSF. The collective dose to unload one transportation cask into the IFSF was calculated to be 0.32 person-rem. This dose is considered representative of a generic dry vault storage facility.

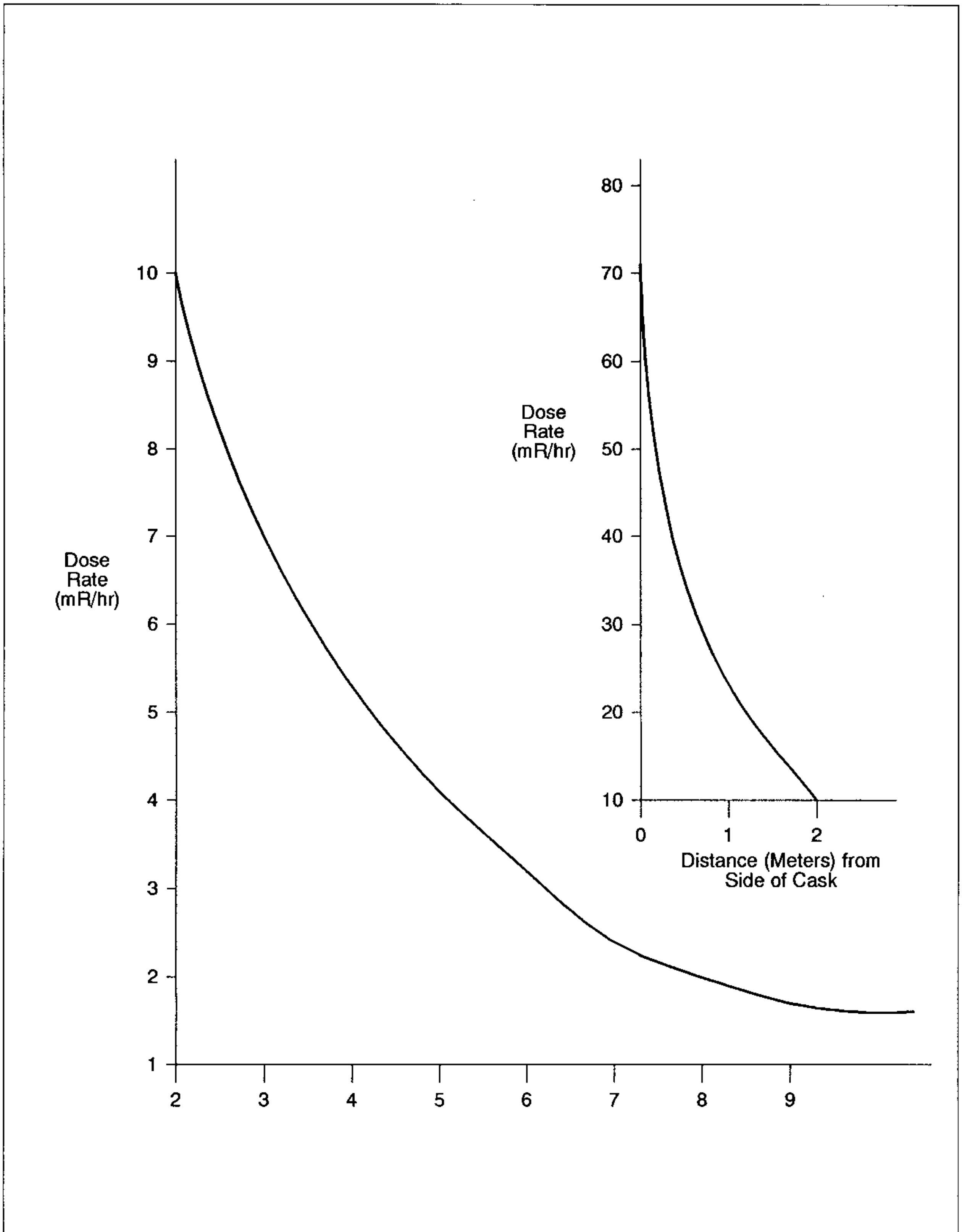


Figure F-50 Bounding Transportation Cask
(Dose Rate at a Distance Normalized to 10 millirem/hour at 2 meters [6.6 feet])

Table F-101 Actual Foreign Research Reactor Spent Nuclear Fuel Transportation Cask Dose Rate Measurements

Date	Cask Model	Fuel	Side Cask Measured Dose Rate (mrem/hr) at 1 m (3.3 ft)
Savannah River Site-Provided Data			
10/2/94	IU04-PEGASE		0.4
10/2/94	PEGASE		2.08
10/2/94	TN-7		8.4
10/2/94	GNS-11		1.2
12/18/87	PEGASE	DR3	1.5
1/26/89	PEGASE	DR3	0.6
12/31/87	PEGASE	ORPHEE	1.4
10/30/86	PEGASE	ORPHEE	0.5
9/1/87	PEGASE	SILOE	1.0
11/5/87	PEGASE	SILOE	0.3
12/30/87	PEGASE	SILOE	0.3
2/7/89	TN-7/2	RHF	15.0
8/16/88	TN-7/2	RHF	20.0
9/25/81	SWED.R2-B/23	AAR	0.9
1/20/89	TN-1	HFR	6.0
7/20/88	TN-1	HFR	0.5
6/8/88	GNS-11	FRJ-2	8.0
8/30/88	GNS-11	FRJ-2	0.8
8/30/88	GNS-11	FRJ-2	10.0
2/27/86	GOSLAR NO.1	FRG	2.0
11/12/86	GOSLAR NO.1	FRG	1.0
2/26/86	GOSLAR NO.2	FRG	0.2
3/10/80	GOSLAR NO.1	ASTRA	0.2
3/14/80	GOSLAR NO.2	ASTRA	0.1
1/29/86	BMI-1	RINC	0.5
2/6/86	BMI-1	RINC	0.4
6/5/84	BMI-1	U.VA.	0.5
6/11/84	BMI-1	U.VA.	0.5
9/11/84	BMI-1	U.MICH.	0.1
10/13/87	BMI-1	U.MICH.	0.1
7/14/81	BMI-1	U.MICH.	1.0
10/6/87	BMI-1	U.MICH.	0.1
Idaho National Engineering Laboratory-Provided Data			
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	BERKLEY-TRIGA	0.5
	BMI-1	MICHIGAN-TRIGA	1.0
	BMI-1	BERKLEY-TRIGA	0.5
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	BNL-HFBR	1.0
	GE-700	U. OF MISSOURI	0.1
	BMI-1	CORNELL-TRIGA	3.5
	BMI-1	MICHIGAN-TRIGA	0.1
	BMI-1	HANFORD-TRIGA	1.0
	BMI-1	HANFORD-TRIGA	1.0

Table F-102 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in Wet Storage

Exposed Workers	*A. Exposure Distance (m)*	*B. Dose Rate (mrem/hr)*	*C. Exposure Time (minutes/cask)*	*D. Dose/Cask-Person (mrem)*	*E. Number of Exposed Workers*	*F. Collective Dose (Person-rem)*
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E+01	1	5.0E-04
	5.0	4.2	10	7.0E+01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Move Cask						
Equipment Operators	0.3	50.0	5	4.2E+00	1	4.2E-03
Subtotal			5	4.2E+00		4.2E-03
Removal of Cask from Transport						
Foreman	8.00	2.1	45	1.6E+00	1	1.6E-03
	2.00	10.0	10	1.7E+00	1	1.7E-03
	1.00	23.0	5	1.9E+00	1	1.9E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	8.00	2.1	30	1.1E+00	1	1.1E-03
	2.00	10.0	20	3.3E+00	1	3.3E-03
	1.00	23.0	10	3.8E+00	1	3.8E-03
Subtotal			60	8.2E+00		8.2E-03
Equipment Operators	4.00	5.3	59	5.2E+00	2	1.0E-02
	1.00	23.0	1	3.8E-01	1	3.8E-04
Subtotal			60	5.6E+00		1.1E-02

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Testing & Verification of Integrity						
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.3	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask to Unloading Pool and Immersion						
Operators	0.30	50.0	2	1.7E+00	3	5.0E-03
	NA	0.1	58	9.7E-02	2	1.9E-04
Subtotal			60	1.8E+00		5.2E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Safeguards	NA	0.1	240	4.0E-01	1	4.0E-04
Subtotal			240	4.0E-01		4.0E-04
Operators	NA	0.1	240	4.0E-01	5	2.0E-03
Subtotal			240	4.0E-01		2.0E-03
Removal of Cask from Unloading Pool						
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and all subsequent operators assumed to be no exposure greater than background						
	NA	0.0	90	0	5	0
Subtotal				0		0
Total				**4.4E+01 (Max.)**		**3.1E-01**

Table F-103 Worker Dose Assessment for Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel in a Dry Storage Facility (Irradiated Fuel Storage Facility) or Generic Vault

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Transport Receipt						
Shipping Agent	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00		1.1E-03
Health Physics Tech	1.0	23.0	5	1.9E+00	1	1.9E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			15	3.6E+00		3.6E-03
Guards	8.0	2.1	30	1.1E+00	1	1.1E-03
Subtotal			30	1.1E+00	1	1.1E-03
Remove Container Cover						
Foreman	10.0	1.5	20	5.0E-01	1	5.0E-04
	6.0	1.2	10	7.0E-01	1	7.0E-04
Subtotal			30	1.2E+00		1.2E-03
Operators	0.3	50.0	15	1.3E+01	3	3.8E-02
	0.3	50.0	10	8.3E+00	1	8.3E-03

Exposed Workers	*A. Exposure Distance (m)*	*B. Dose Rate (mrem/hr)*	*C. Exposure Time (minutes/cask)*	*D. Dose/Cask-Person (mrem)*	*E. Number of Exposed Workers*	*F. Collective Dose (Person-rem)*
	1.0	23.0	1	3.8E-01	1	3.8E-04
	2.0	10.0	10	1.7E+00	2	3.3E-03
Subtotal			36	2.3E+01		5.0E-02
Survey Cask						
Health Physics Tech	0.3	50.0	45	3.8E+01	1	3.8E-02
Subtotal			45	3.8E+01		3.8E-02
Removal of Impact Limiters						
Foreman	5.0	4.2	50	3.5E+00	1	3.5E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	5.0	4.2	45	3.2E+00	1	3.2E-03
	0.3	50.0	15	1.3E+01	1	1.3E-02
Subtotal			60	1.6E+01		1.6E-02
Operators	0.3	50.0	10	8.3E+00	3	2.5E-02
	0.5	36.0	60	3.6E+01	2	7.2E-02
Subtotal			70	4.4E+01		9.7E-02
Removal of Cask from Transport to Transfer Cart						
Foreman	8.0	2.1	45	1.6E+00	1	1.6E-03
	2.0	10.0	10	1.7E+00	1	1.7E-03
	1.0	23.0	5	1.9E+00	1	1.9E-03
Subtotal			60	5.2E+00		5.2E-03
Health Physics Tech	8.00	2.1	30	1.1E+00	1	1.1E-03
	2.00	10.0	20	3.3E+00	1	3.3E-03
	1.00	23.0	10	3.8E+00	1	3.8E-03
Subtotal			80	8.2E+00		8.2E-03
Equipment Operators	4.00	5.3	59	5.2E+00	2	1.0E02
	1.00	23.0	1	3.8E-01	1	3.8E-04
Subtotal			60	5.6E+00		1.1E-02
Testing & Verification of Integrity, Lid Bolt Removal						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Health Physics Tech	5.00	4.2	30	2.1E+00	1	2.1E-03
	2.00	10.0	28	4.7E+00	1	4.7E-03
	0.30	50.0	2	1.7E+00	1	1.7E-03
Subtotal			60	8.4E+00		8.4E-03
Operators	0.30	50.0	30	2.5E+01	2	5.0E-02
	4.00	5.3	30	2.7E+00	2	5.3E-03
Subtotal			60	2.8E+01		5.5E-02
Movement of Cask into Handling Cove						
Foreman	4.00	5.3	60	5.3E+00	1	5.3E-03
Subtotal			60	5.3E+00		5.3E-03
Operators	4.0	5.3	10	8.8E-01	2	1.8E-03
	NA	0.1	50	8.3E-02	2	1.7E-04
Subtotal			60	9.7E-01		1.9E-03
Cask Unloading/Inspection/Storage						
Foreman	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04
QA Inspector	NA	0.1	480	8.0E-01	1	8.0E-04
Subtotal			480	8.0E-01		8.0E-04

Exposed Workers	A. Exposure Distance (m)	B. Dose Rate (mrem/hr)	C. Exposure Time (minutes/cask)	D. Dose/Cask-Person (mrem)	E. Number of Exposed Workers	F. Collective Dose (Person-rem)
Operators	NA	0.1	480	8.0E-01	2	1.6E-03
Subtotal			480	8.0E-01		1.6E-03
Removal of Cask from Handling Cove						
Foreman	NA	0.1	60	1.0E-01	1	1.0E-04
Subtotal			60	1.0E-01		1.0E-04
Operators	NA	0.1	60	1.0E-01	2	2.0E-04
Subtotal			60	1.0E-01		2.0E-04
Replacement of Cask of Transport and All Subsequent Operators Assumed to be No Exposure Greater than Background	NA	0.0	90	0	5	0
Subtotal			90	0		0
Total				**4.4E+01 (Max.)**		**3.2E-01**

DSC = Dry Shielded Canister

Transfer Between Storage Facilities: The collective doses were calculated for loading fuel into a pod, a dry vault (i.e., the IFSF), and dry cask (i.e., Calvert Cliffs NUHOMS) or during transfer between these facilities. It was assumed that larger commercial spent nuclear fuel transportation casks are used for intersite and intrasite movement of foreign research reactor spent nuclear fuel within the United States. Their capacity is approximately four times that of the foreign research reactor spent nuclear fuel transportation casks from overseas. It was also assumed that the transfer cask for the dry cask design has a capacity which is approximately eight times that of the overseas foreign research reactor spent nuclear fuel transportation casks.

F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Appendix F.6 describes only the methodologies and assumptions used for estimating radiation exposure (doses) to individuals and the general public from releases of radioactivity during incident-free operations and hypothetical accidents at potential management sites. The descriptions of similar evaluations for ground and marine transportation and port accidents are documented in Appendix E and Appendix D.

F.6.1 Analysis Methods for Evaluation of Radiation Exposure

F.6.1.1 General

An evaluation of incident-free operations and hypothetical accidental radioactive material releases at the proposed storage sites was performed to assess the impact of possible radiation exposure to individuals and the general population. The analysis assumes that the same operations are being carried out at different potential storage locations. The impact of the same radioactive material releases was evaluated at all potential sites. This approach provides a consistent method for comparing the effects of the proposed alternative actions.

F.6.1.2 Exposure Impacts to Be Estimated

The impact of radiation exposure (dose) to the following individuals and the general population is calculated for incident-free operation of the spent nuclear fuel storage facility and for accident conditions:

- ***Worker:*** An individual located 100 m (330 ft) from the radioactive material release point.[2] The dose to the worker is calculated for the 50th-percentile meteorology only (DOE, 1992a).

- ***MEI:*** A theoretical individual living at the storage site boundary and receiving the maximum exposure.

- ***NPAI:*** At some storage sites, highways used by the public may cross the Federal reservation where foreign research reactor spent nuclear fuel operations could be conducted. Consequently, these analyses included evaluation of the exposure to a theoretical motorist who might be stranded on such a highway at the time of an accident. Based on experience from emergency exercises, emergency response teams would be able to evacuate such an individual within 2 hours, so this was the exposure time used in the calculations.

- ***General population*** within an 80 km (50 mi) radius of the facility.

The doses to the NPAI, MEI and general population are calculated for the 50th- and 95th-percentile meteorological conditions. The details of exposure times for MEI, NPAI, worker, and general public are given in Section F.6.4.1.

The radiation dose to individuals and the public resulting from exposure to radioactive contamination was calculated using the following potential pathways:

- external direct exposure from immersion in the airborne radioactive material (air immersion),

- external direct exposure from radioactive material deposited on the ground (ground surface),

- internal exposure from inhalation of radioactive aerosols and suspended particles (inhalation), and

- internal exposure from ingestion of contaminated terrestrial food and animal products (ingestion).

The radiation dose is estimated by the GENII (Version 1.485) computer program (Napier et al., 1988) in a manner recommended by the International Commission on Radiological Protection in Publications 26 and 30 (ICRP, 1977; ICRP, 1979-1982). Committed dose equivalents [3] are calculated for organs such as the gonads, breasts, red bone marrow, lungs, thyroid, bone surface, liver, lower large intestine, upper large

---

2 *For elevated release, the worker dose was calculated at a point of maximum dose. The distance at which the maximum dose occurs is frequently greater than 100 m (300 ft) for elevated release.*

3 *The definitions of committed dose equivalents, committed effective dose equivalents, and total effective dose equivalents are consistent with those given in 10 CFR Part 835, "Occupational Radiation Protection; Final Rule," (DOE, 1993a).*

intestine, small intestine, and stomach. Weighting factors are used for various body organs to calculate weighted or committed effective dose equivalent (EDE) from radiation inside the body due to inhalation or ingestion. The committed EDE value is the summation of the committed dose equivalent to the specific organ weighted by the relative risk to that organ compared to an equivalent whole-body exposure.

The program also estimates deep-dose equivalent for the external exposure pathways (immersion in the radioactive material and exposure to ground contamination) and a 50-year committed EDE for the internal exposure pathways. The sum of the deep-dose equivalent for external pathways and the committed EDE for internal pathways is called the cumulative dose or "total EDE" in this EIS and is also estimated by the GENII program.

The exposure from ingestion of contaminated terrestrial food and animal products is calculated on a yearly basis. However, it is expected that continued consumption of contaminated food products by the public would be suspended if the projected dose exceeds the protective action guidelines for use in the event of radiological accidents (EPA, 1991). No reduction of exposure due to protective actions was accounted for in this analysis, however. This results in a conservative approach that may overestimate health effects within an exposed population, but allows for consistent comparisons between alternatives.

F.6.1.3 Evaluation of Health Effects

Health effects are calculated from the exposure results. The risk factors used for calculations of health effects are taken from International Commission on Radiological Protection Publication 60 (ICRP, 1991) (see Table F-104). From this list only the factors associated with the fatal cancers were used in the analysis. Other factors are given as additional information for completeness.

Table F-104 Risk Estimators for Health Effects from Ionizing Radiation

Effect	*Risk Factor (probability/rem)*[a]	
	Worker	*General Population*
Fatal cancer (all organs)	4.0×10^{-4}	5.0×10^{-4}
Weighted nonfatal cancer[b]	8.0×10^{-5}	1.0×10^{-4}
Weighted genetic effects[b]	8.0×10^{-5}	1.3×10^{-4}

[a] *For high individual exposures (20 rem), the risk factors are multiplied by a factor of two.*

[b] *In determining a means of assessing health effects from radiation exposure, the International Commission on Radiological Protection has developed a weighting method for nonfatal cancers and genetic effects. These factors are provided here for information only and were not used in this analysis. Genetic effects can only be applied to population, not to individuals.*

F.6.1.4 Population

Population distributions specific to each site were used for the evaluations. The population distributions were obtained from each site. The population information was obtained in 16 compass directions and 10 radial distances from the likely location of a foreign research reactor spent nuclear fuel storage facility to an 80 km (50 mi) total distance.

F.6.1.5 Meteorology

Meteorology specific to each site was used in the evaluation. The site-specific meteorological data was prepared, or acquired from each candidate storage site, in the form of joint frequency distribution in terms of percentage of time that the wind blows in specific directions (i.e., south, south-southwest, southwest, etc.) for the given midpoint (or average) wind speed class and atmospheric stability. Accident consequence calculations were performed using 50th- and 95th-percentile meteorological conditions. The 50th-percentile condition represents the median meteorological condition, and is defined as that for which more severe conditions occur 50 percent of the time. The 95th-percentile condition represents relatively low probability meteorological conditions which produce higher calculated exposures, and is defined as that condition which is not exceeded more than 5 percent of the time. GENII determines 50th- and 95th-percentile meteorological conditions using site-specific joint frequency distribution weather data.

F.6.1.6 Computer Programs

The following computer programs were used to evaluate the radiation exposure to the specified individuals and the general population.

GENII: The GENII code (Napier et al., 1988) was used to model both acute and chronic releases to the atmosphere. This code was developed by the Pacific Northwest Laboratory to incorporate the internal dosimetry models recommended in International Commission on Radiological Protection Publication 26 (ICRP, 1977) and Publication 30 (ICRP, 1979-1982) into environmental pathway analysis models in use at the Pacific Northwest Laboratory. This code has been used by the Pacific Northwest Laboratory and other laboratories in site-wide dosimetry calculations. It has been extensively validated and quality assured.

ORIGEN2: ORIGEN2 (Croff, 1980) is a computer code system for calculating the buildup and decay of radioactive materials (fission products, actinides, and activation products). The code input was modeled to describe the HEU and low enriched uranium (LEU) research reactor nuclear fuel system and used neutronic cross-section data that are distinct to these fuels. ORIGEN2 has been used extensively by the Argonne National Laboratory in the RERTR program in estimating nuclide inventories of irradiated fuels. The code and the specific neutronic cross-section parameters for HEU and LEU fuels were acquired from the Argonne National Laboratory. ORIGEN2 is widely used and accepted throughout the nuclear industry.

F.6.2 Screening/Selection of Accidents for Detailed Examination

Accidents considered for inclusion in the detailed analyses are similar to those analyzed in the Programmatic SNF&INEL Final EIS for the spent nuclear fuel storage facility operations (DOE, 1995g). The analyzed accident scenarios in the Programmatic SNF&INEL Final EIS for each potential storage site were reviewed to identify the bounding accidents to be considered in this EIS. The review included accidents initiated by natural phenomena (earthquakes, tornadoes, hurricanes, etc.) and accidents initiated from human or equipment failure (fires, explosions, aircraft crashes, transportation accidents, and terrorism).

A review of accidents indicates that only severe accident conditions could result in a release of radioactive material to the environment or an increase in radiation levels. Some types of accidents, such as procedure violations, spills of small volumes of water containing radioactive particles, and most other types of common human error may occur more frequently than the more severe accidents analyzed. However, these accidents do not involve enough radioactive material or radiation to result in a significant release to the environment or a meaningful increase in radiation levels. Stated another way, the very low consequences associated with these events produce smaller risks than those for the accidents analyzed, even when combined with a higher probability of occurrence. Consequently, they have not been included in the results presented in this EIS.

Accidents initiated at nearby facilities, either by other activities unrelated to spent nuclear fuel handling or storage or during construction of a wet or dry storage facility, would not produce effects more severe than the sequence of events being analyzed. This is because foreign research reactor spent nuclear fuel undergoing examination or in the process of being stored would not need special conditions or uninterrupted operator attention to prevent overheating or to maintain containment or shielding. Therefore, evacuation in response to an accident at some other facility would not compromise integrity of the spent nuclear fuel.

The potential for common-cause accidents at a storage facility has been considered. It is possible for natural phenomena, like an earthquake, to produce more than one accident at a site causing a situation that results in a release of radioactive material into the atmosphere or an increase in radiation levels due to loss of shielding. However, the probability of two or more accidents having maximum consequences occurring concurrently is less than the probability of the individual events. For example, if an earthquake affected the wet storage facility, a crane might fail causing damage to stored spent nuclear fuel, and the water pool might drain. The impacts for this could be conservatively estimated by summing the consequences. Similarly, consequences from spent nuclear fuel facilities within a DOE site could be combined to conservatively estimate site-wide impacts. But again, the probability of a common-cause event resulting in this number of consequences is lower than the probability of individual accidents because, due to separation distances, the severity of impact will vary between facilities. The existing security measures in effect at the management sites would essentially preclude any sabotage or terrorist activity. Further, any acts of terrorism are expected to result in consequences which are bounded by the results of accidents analyzed. Thus, no specific analyses of the results of terrorist acts were conducted.

Based on the above, the review identified the following bounding accident scenarios:

- criticality caused by human error during operation, equipment failure, or earthquake;
- mechanical damage to foreign research reactor spent nuclear fuel during examination and preparation (cropping off the aluminum and nonfuel end of a fuel); and
- accident involving an impact by either an internal or external initiator with and without an ensuing fire.

F.6.3 Accident Scenarios Considered

A total of six bounding accident scenarios for the handling and storage of foreign research reactor spent nuclear fuel were identified for detailed analysis. Each of these accident scenarios was evaluated at each storage location using identical source terms. As described below, three of the bounding accident scenarios apply to wet storage and three apply to dry storage.

F.6.3.1 Wet Storage Bounding Accident Scenarios

Three hypothetical accident scenarios were evaluated for foreign research reactor spent nuclear fuel stored in water pools: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage due to operator error, (b) an accidental criticality, and (c) an aircraft crash into the water pool facility. In addition to these three scenarios, a dropped fuel cask was also considered to be a foreseeable accident. However, as will be seen in Section F.6.4.4.4, the consequences of this accident are bounded by the cutting into a fuel region scenario. Therefore, a dropped fuel cask was not evaluated in detail.

F.6.3.2 Dry Storage Bounding Accident Scenarios

Three hypothetical accidents were evaluated for foreign research reactor spent nuclear fuel handled in dry storage: (a) fuel element breach (i.e., cutting into the fuel region) or mechanical damage during examination work and handling, (b) dropping of a fuel cask, and (c) an aircraft crash with ensuing fire in the dry storage facility. No credible mechanism was identified for an accident criticality in dry storage.

F.6.4 Bounding Accident Evaluation

F.6.4.1 Basic Assumptions

The analysis of airborne releases from hypothetical accidents is performed using the GENII Version 1.485 computer program. Unless otherwise stated, the following conditions were used when performing calculations. In most cases, these are the default conditions in the GENII program.

Meteorological Data:

- Fiftieth- and 95th-percentile meteorological conditions for each storage site were defined using site-specific joint frequency distribution weather data.
- The release is assumed to occur at ground level (0 m).
- Mixing layer height is 1,000 m (3,280 ft). Airborne materials freely diffuse in the atmosphere near ground level in what is known as the mixing depth. A stable layer exists above the mixing depth which restricts vertical diffusion above 1,000 m.
- Wet deposition is zero (it is assumed that no rain occurs to accelerate deposition and reduce the size of area affected by the release).
- Dry deposition of the cloud is modeled. During movement of the radioactive plume, a fraction of the radioactive material in the plume is deposited on the ground due to gravitational forces. The deposited material no longer contributes to the air immersion dose from the plume, but now contributes as exposure from ground surface radiation and ingestion.
- The quantity of deposited radioactive material is proportional to the material particle size and deposition velocities (in m/sec) used in the GENII code as follows:

solids = 0.001	halogens = 0.01	noble gases = 0.0
cesium = 0.001	ruthenium = 0.001	

- If radioactive releases occur through a stack, then additional plume dispersion can be accounted for by considering the beneficial effects of jet plume rise. In this analysis, jet plume rise is ignored.
- When released gases have a heat content, the plume can disperse more quickly. In this calculation, buoyant plume effects are ignored.

Inhalation Data:

- Breathing rate is 330 cm^3/sec (20.1 in^3/sec) for the worker and the NPAI; 270 cm^3/sec (16.5 in^3/sec) for people at the site boundary and beyond (the MEI and the general population).
- Particle size is 1.0 micro-meter (micron).
- The internal exposure period is 50 years for the individual organs and tissues evaluated.
- Exposure during passage of the entire plume is assessed for the MEI and the general public. Exposures to the worker and NPAI are discussed below.
- Inhalation exposure factors are based on International Commission on Radiological Protection Publication 30 (ICRP, 1979-1982).

Mitigating Factors:

For the MEI and members of the general public residing at the site boundary and beyond, no allowances are made for any preventive or mitigative actions that would limit their exposure. These individuals are assumed to be exposed to the contaminated plume during the entire period of its passage, as it travels downwind from the accident site. Similarly, no action is taken to prevent these people from continuing their normal daily routine, including ingestion of the potentially contaminated terrestrial food and animal products. It is assumed, however, that the public would spend approximately 30 percent (about 8 hours) of the day within their homes or other buildings. Therefore, the exposure of the general public to radiation from contaminated ground surface is reduced appropriately. Calculations were done on a yearly basis to determine the effective annual dosage from inhalation, external exposure, and ingestion, and an associated dose commitment extending over a 50-year period from initiation of intake (NRC, 1977a).

Onsite workers would be trained to take quick, decisive action during an accident. These individuals would be trained to quickly evacuate the affected area and move to well-defined "relocation" areas on the facility. Therefore, it is assumed that workers would be exposed to only 5 minutes of the radioactive plume as they move to relocation centers. Once the plume has moved offsite and downwind, the workers would be instructed to walk to vehicles waiting to evacuate them from the site. It is assumed that an additional 15 minutes would be required to evacuate the workers from the contaminated area and, therefore, the workers would receive a total of 20 minutes of exposure to radioactive material deposited on the ground. No ingestion of contaminated foods is assumed for these individuals.

Individuals that may be traversing the site in a vehicle (i.e., NPAI) would be evacuated from the affected area within 2 hours. This is based on the availability of security personnel at all locations to oversee the removal of collocated workers and travelers in a safe and efficient manner. Therefore collocated workers and travelers would be exposed to the entire contaminated plume as it travels downwind for a period not to exceed 2 hours. Similarly, the radiation from the deposited radioactive materials would be limited to a 2-hour period. No ingestion of contaminated foods is assumed for these individuals.

Table F-105 provides the individual exposure times used in the accident analyses presented later in this appendix.

Table F-105 Estimated Individual Exposure Times

Exposure Type	*Worker (100 m)*	*NPAI*	*MEI/General Public*
To Plume	5 min	100% of release time up to 120 min	100% of release time
To Fallout on Ground Surface	20 min	120 min	0.70 yr
To Food	NA	NA	1 yr

F.6.4.2 Source Term

The source term is the amount of respirable radioactive material, in terms of Ci (curies), that are released to the air. The airborne source term is typically estimated by the following five-component linear equation:

Source Term = MAR x DR x ARF x RF x LPF

where:

MAR = Material-at-Risk (g or Ci),

DR = Damage Ratio,

ARF = Airborne Release Fraction (or Airborne Release Rate for continuous release),

RF = Respirable Fraction, and

LPF = Leak Path Factor.

MAR: The MAR is the amount of radionuclides (in g or Ci of activity for each radionuclide) available to be acted upon by a given physical stress (i.e., an accident). The MAR is specific to a given process in the facility of interest. It is not necessarily the total quantity of material present, but is that amount of material in the scenario of interest postulated to be available for release.

DR: This is the fraction of material exposed to the effects of the energy/force/stress generated by the postulated event. For the bounding accident scenarios discussed in this document, the value of DR is assumed to be one (i.e., all exposed material is released), unless otherwise specified.

ARF: This is the fraction of the material that becomes airborne due to the accident. Generic ARF values from DOE sources (Elder et al., 1986; DOE, 1994d) are used in this document unless other values more appropriate to a particular accident scenario are used for ARF. The values for ARF are summarized in Table F-106.

RF: This is the fraction of the material, with particle sizes of 10 micro-meters (microns) or less (DOE, 1994d) that could be retained in the respiratory system following inhalation. The term RF is applied only for the inhalation pathway.

LPF: The LPF accounts for the action of removal mechanisms, such as containment systems, filtration, deposition, etc., to reduce the amount of airborne radioactivity that is ultimately released to occupied spaces of the facility or to the environment. An LPF of 1.0 (i.e., no reduction) is assigned in accident scenarios involving a major failure of confinement barriers.

Table F-106 Release Fractions[a] for Various Release Mechanisms

Material	Release Mechanisms		
	Fuel breach[a]	Fire	Criticality Accident[b]
Gas			1.0
Noble Gas	1.0	1.0	
Krypton	0.3	1.0	
Other Noble Gas	0.1	1.0	
Halogens	0.1	1.0	0.25[d]
Iodine-129	0.25		
Solids			
Volatile	0.01[c]	2.5×10^{-4} [d,e]	
Nonvolatile	0.01	2.5×10^{-6} [d,e]	

Source: DOE, 1995g

[a] *As recommended in Elder et al., 1986.*

[b] *Regulatory Guide values (NRC, 1977b, 1979b, and 1988b).*

[c] *Actually semi-volatile (cesium, rhodium, antimony, selenium, technetium, and tellurium); review on a case-by-case basis.*

[d] *Includes release fraction, respirable fraction and plate-out.*

[e] *Data from DOE, 1995g.*

F.6.4.3 Description of Radiological Accident Scenarios and Generic Parameters

As discussed previously, the accident screening and selection process led to selection of six bounding accident scenarios involving radioactive materials. Appropriate assumptions also have been discussed regarding meteorological parameters, dispersion parameters, dose estimates, and emergency response and protective actions. Each of the accident scenarios is described in the following text according to the major headings listed below:

- Description of Accident,
- Development of Radioactive Source Term, and
- Dose Calculations and Results.

The contents of these sections and a summary of the generic parameters used follow.

Description of Accident provides a basis for accident selection and discusses possible initiating events. A qualitative assessment of scenario likelihood is provided.

Development of Radioactive Source Term describes the assumptions that apply to the development of the resulting source term. Specifically, it discusses the various multipliers (defined earlier in this section) that convert the MAR to the source term.

These multipliers have the following values:

- DR is 1.0, unless otherwise specified.
- ARF is taken from Table F-106, or clearly stated if different.
- LPF is 1.0 for a major failure of confinement barriers.

Dose Calculations and Results relates the computer modeling to the specific accident scenario, and documents the results. Specifically, these subsections accomplish the following:

- describe assumptions and unique input parameters (other than the source term) used in the computer model,
- document the computer model output in terms of exposure to radionuclides for individuals and for the general population within a 80 km (50 mi) radius, and
- assess the potential for health effects.

Unless otherwise specified, the meteorological/dispersion parameters and estimated exposure times summarized are used in the dosimetry calculations for specific accident scenarios. Under some circumstances, facility worker exposures could be either greater or less than these nominal values.

F.6.4.4 Accident Scenario Descriptions and Source Terms

F.6.4.4.1 Fuel Element Breach

Description of Conditions: Fuel element mechanical damage due to handling during examination, such as accidentally cutting into the fuel region, was assessed. This hypothetical accident results from inadvertent cutting across the fuel region when cropping off the aluminum and nonfuel ends of a fuel unit. All noble gas isotopes are postulated to be released to the facility building and escape to the environment. The majority of the volatile and solid nuclides are likely to be retained in the fuel or the facility exhaust filters. The resulting airborne release to the environment was evaluated.

Likelihood: The frequency of this scenario is estimated to be 0.16/yr (DOE, 1995g). This frequency estimate is based on historical operation data (one event in 6 years) for a spent nuclear fuel storage facility. This estimate is conservative for the case of foreign research reactor spent nuclear fuel storage because the majority of the spent nuclear fuel elements are expected to be cropped prior to their emplacement in a transportation cask at a foreign research reactor. Nevertheless, this estimate is retained for the evaluation of the potential risk associated with the handling and preparation of foreign research reactor spent nuclear fuel for storage in both a dry and a wet storage facility.

Source Term: Conditions used in developing the source term are as follows:

- Only one spent nuclear fuel element is damaged. This is because only one spent nuclear fuel element is being handled at a time.

If the spent nuclear fuel cutting accident occurs in a dry cell (dry storage), the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the atmosphere. Here, it was assumed that all noble gases in an irradiated fuel element would be released. This is conservative, since foreign research reactor fuels are dispersion fuels in which the gaseous fission products are essentially trapped within the fuel matrix. This is different than for commercial reactor fuel, where gaseous fission products collect in the gap between the fuel and its sealed metal fuel rod and are readily released if the rod is damaged.

- Twenty-five percent of the halogens in the spent nuclear fuel are released to the environment. This is also conservative for the reason stated above.
- One percent of the particulate fission products is released to the dry cell from the spent nuclear fuel element, and 99.9 percent removed prior to release to the environment by the normally installed high-efficiency particulate air filters. The use of 99.9 percent efficiency is conservative, since normal efficiency of installed high-efficiency particulate air filters is greater than 99.99 percent.
- Cesium (Cs) and Ruthenium (Ru) behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

If the spent nuclear fuel cutting accident occurs under water (wet storage) the following assumptions apply:

- All (100 percent) of the noble gases available for release are released to the environment.
- Twenty-five percent of the halogens available for release will be released to the pool, and only 10 percent of this amount will be released to the air. This additional reduction is due to the fact that halogen gases dissolve in the water as they escape (leak out) from the failed fuel. Based on solubility alone, it is expected that all iodines are dissolved in the water pool before they get to the pool surface. In spite of this fact, for the purposes of the analyses, it was assumed that 2.5 percent of halogens available for release will be released to the atmosphere.
- There is no particulate fission product release to the environment. All particulates are retained in the pool water.
- Since only gaseous fission products are released to the air inside the facility, installed high-efficiency particulate air filters would not provide additional reduction in the amount of material released to the environment.
- The release to the environment occurs at a constant rate over a 15-minute period.

F.6.4.4.2 Accidental Criticality

Description of Conditions: In this hypothetical accident scenario, an accidental uncontrolled chain reaction producing 1×10^{19} fissions is postulated. The 10^{19} fission criticality is a very conservative assumption for the spent nuclear fuel pool. This assumption is only applicable to liquid processes (such as uranium reprocessing) as stated in Regulatory Guides 3.33 and 3.34 (NRC, 1979a and 1979b). This criticality is assumed to consist of an initial burst of 10^{18} fissions in 0.5 seconds, followed at 10 minute intervals for the next 8 hours by a burst of 2×10^{17} fissions, for a total of 10^{19} fissions. The total yield for a moderated solid system, as applicable to the spent nuclear fuel in a wet pool, is estimated to be on the order of 10^{18} fissions. This is because the initial criticality will disrupt the critical geometry and no further criticality burst will occur.

The criticality occurs in the water pool and the spent nuclear fuel remains covered in the water. The fission products released include those specified in Regulatory Guide 3.34 (NRC, 1979b) from the criticality over an 8-hour period, plus fission products existing in the fuel as a result of its original use in the foreign research reactor. Removal of fission products by the pool water is considered in the analysis.

Criticality is not considered in the dry storage because the licensing design basis for spent nuclear fuel dry storage design facilities precludes the consideration of any criticality accident by design. The design must demonstrate, through rigorous structural and criticality analyses, that the likelihood of a criticality is incredible or unforeseeable. No effective moderator, such as water, exists in a dry storage design; and, even if flooded, it remains subcritical.

***Likelihood*:** The frequency of this scenario is estimated at 3.1 x 10^{-3} per year (DOE, 1995g). The estimation of this frequency was conservatively based on a statistical evaluation considering that no accidental criticality event with spent nuclear fuel storage has occurred (DuPont, 1983b). This frequency is estimated by considering both the various process-related upset conditions and the natural phenomena hazard (i.e., earthquake and tornadoes) initiated criticality events. The magnitude of fission yield for such a criticality accident was estimated to range from about 5 x 1017 to 1 x 10^{19} fissions. The historical criticality accidents at different DOE facilities dealing with spent nuclear fuels indicate a much smaller fission yield than that evaluated here. The frequency of an accidental criticality of the magnitude evaluated here is estimated to be between one and two orders of magnitude less than the estimated frequency.

***Source Term*:** Conditions used in developing the source term are as follows:

- The fractions of the fission products from damaged spent nuclear fuel elements released to the building are 100 percent of the noble gases, 25 percent of the halogens, 0.1 percent of the Ru, and 0.05 percent of the Cs and remaining solids (NRC, 1977b, 1979b, and 1988b).
- Fission products from 10 spent nuclear fuel elements damaged in the criticality accident are also released in addition to the gaseous fission products created by the criticality event.
- A high-efficiency particulate air filter removes 99.9 percent of the solid fission products that were released to the air inside the facility before they enter the environment.
- The release to the environment occurs at a constant rate over a 15-minute period. This is conservative as compared to the 8-hour release allowed in Regulatory Guide 3.34 (NRC, 1979b).

F.6.4.4.3 Aircraft Crash

Dry Storage:

***Description of Conditions*:** A hypothetical aircraft accident scenario was developed for the dry storage option. This accident is analyzed only at storage sites that have a likelihood of accident occurrence greater than 10^{-7} per year. The consequences of this accident are expected to bound all other dry storage accident scenarios involving an impact that results in fire. The aircraft crash accident is postulated to cause damage to a single transfer container in the dry unloading cell in a modular vault storage facility. Engineering experience indicates that most of the aircraft structure is stopped by the dry storage building structure. Only a heavy dense jet engine rotor shaft is expected to be capable of penetrating the building and damaging the container. Due to the severity of the impact, it was assumed that the cask is breached and the fuel elements in the cask are damaged. The release of fission products occurs due to the impact and resultant fire (i.e., from aviation fuel).

The accident scenario for a dry cask storage facility is similar to that of a modular vault facility. The aircraft crash analysis is the only accident scenario applicable to a dry cask storage. In this scenario, it is expected that the concrete structure which houses the storage canisters is sufficiently rugged that it can survive an aircraft accident with no significant damage to the spent nuclear fuel.

***Likelihood*:** The frequency of this scenario is site dependent. DOE, as part of the Programmatic SNF&INEL Final EIS, has performed calculations of aircraft crash hit frequencies at potential storage sites (i.e., Savannah River Site, Idaho National Engineering Laboratory, Oak Ridge Reservation, Hanford Site, and Nevada Test Site) for naval fuel (DOE, 1995g). The reported crash frequencies are: 2 x 10^{-6} per year for the Savannah River Site, 1 x 10^{-6} per year for the Oak Ridge Reservation, 4 x 10^{-7} per year for Nevada Test Site, 7 x 10^{-8} per year for the Idaho National Engineering Laboratory, and 4 x 10^{-8} per year for the Hanford Site. These frequency estimates were based on the number of commercial air carriers and military aircraft passing within a 10-mile radius of the proposed storage location at these sites. The calculations for the Idaho National Engineering Laboratory also included potential hazards from a nearby airport. These calculations were performed very conservatively, by considering that all the overflights within the 10-mile radius will pass directly over the storage location at each site.

A new assessment of aircraft impact probabilities for the Idaho National Engineering Laboratory chemical processing plant indicates a frequency of aircraft crash into a dry storage facility the size of the IFSF of about 2.6 x 10^{-10} per year from overflights and 3.5 x 10^{-7} per year from airport-related flights near the plant (WINCO, 1994). (The IFSF effective area is five times that considered in the evaluation for the naval fuel storage area, which represents the critical areas containing spent nuclear fuel. Therefore both results are consistent, from the overall crash frequency point of view at the Idaho National Engineering Laboratory.)

In order to provide an understanding of the rationale used in this EIS for this scenario, an overview of the aircraft crash analysis approach is presented. In general, the aircraft crash hit frequency is calculated based on four factors: number of flight operations (takeoff, landing, overflight), aircraft crash rate, facility effective area, and an assumption of crash area distribution. Several models are currently used to estimate the hit frequency. The results of these models are driven by the assumptions regarding the target area and crash area distribution. For example, assuming that overflights (high or low altitude) pass over the facility inherently assumes that the crash area distribution is a straight line. This overestimates the frequency by at least a factor of 10 (approximate width of an airway). In calculating effective area, the analysis considers that an aircraft can hit a facility either directly (falling on the building, footprint area), by skidding into the building (skid area), or in an angular impact (shadow area). Depending on the assumptions of skid length and the angular approach of a crash terminating aircraft, the sum of the latter two areas may contribute between 80 to 95 percent of the total effective area. It is important to note that aircraft that fall vertically with the greatest impact contribute between 1 and 10 percent to the overall crash rate. Therefore, for the majority of cases, the aircraft will hit the ground before it hits the facility.

Based on the above summary, it is considered that frequencies reported in the Programmatic SNF&INEL Final EIS are conservative by at least a factor of 10 for all sites except the Idaho National Engineering Laboratory. Nonetheless, for the purposes of analyses and consistency, this EIS will consider frequencies similar to those used in the Programmatic SNF&INEL Final EIS. The potential aircraft crash frequency at the Oak Ridge Reservation, the Nevada Test Site, the Idaho National Engineering Laboratory, and the Savannah River Site is conservatively set at 10^{-6} per year. This scenario will not be applicable to the Hanford Site, where the estimated frequency is less than 10^{-7} per year.

***Source Term*:** Conditions used in developing the source term are as follows:

- Only one transfer cask containing 20 spent nuclear fuel elements would be damaged by the impact and the resultant fire. This is based on the fact that, if an aircraft hits the building, only the transfer cask is susceptible to damage by the crash. The stored casks are protected by a three-foot concrete shield, and therefore would not be affected by the crash. Based on a conservative estimate of the duration of the transfer operation, the transfer cask could be damaged by the accident only one percent of the time.

- Of the available fission products, 100 percent of the noble gases, 100 percent of the halogens, 2.5 percent of the cesium, and 0.025 percent of the remaining solids are released to the environment. The overall, respirable fractions of fission products released to the environment are consistent with that given in Table F-106 for a fire scenario.
- The release to the environment occurs at a constant rate over a 15-minute period.
- No filtration by high-efficiency particulate air filters is assumed.

For dry cask storage, it was assumed that the ruggedness of the overall dry cask structure is similar to that of a transportation cask. Based on this assumption, the accident source terms were assumed to be similar to that of aluminum-based spent nuclear fuel source terms for the highest severity accident (cask damage and fire) utilized in the RADTRAN accident analysis (DOE, 1995g). The overall source terms for this scenario include: 63 percent of noble gases, 6×10^{-3} percent of halogens, 1×10^{-3} percent of cesium, 2.4×10^{-4} percent of ruthenium, and 1×10^{-5} percent of other solid fission products available in a dry cask.

Wet Storage:

***Description of Conditions*:** Impact into water pools by aircraft with resulting damage to the spent nuclear fuel elements stored inside the pool was evaluated. The hypothetical accident might damage the fuel either by the aircraft directly striking it or by the aircraft causing sufficient damage to the building to cause part of the building to collapse and strike the fuel. Fission products are released from the spent nuclear fuel units into the water pool, however, the pool water is not released to the environment. An aircraft crash into a water pool would not produce enough force to cause the pool to leak because the walls of the water pool are constructed of thick reinforced concrete with earth surrounding them, making them very strong. In addition, based on the discussion provided above, it was judged unlikely that an aircraft would impact the water pool at an angle steep enough to expose the floor of the pool or the walls of the pool below the water level to direct impact.

***Likelihood*:** The same frequency as discussed above will be used for an aircraft crash into a wet storage facility.

***Source Term*:** Conditions used in developing the source term are as follows:

- It was estimated that about 140 spent nuclear fuel elements would damaged. This estimate was based on the consideration of the size of spent nuclear fuel allowing fuel stacking and an assumption that only one percent of the upper stacked fuel will be damaged.
- Of the available fission products, 100 percent of the noble gases and 25 percent of the halogens are released to the pool water. Due to the presence of pool water, a reduction of the halogen release by a factor of 10 occurs prior to release to the environment.
- The pool water is not expected to be lost and the solid fission products from ruptured and damaged fuel elements remain in the water. However, for the purposes of this analysis, it was conservatively assumed that 0.01 percent of the solid fission products (including Cs and Ru) released from the damaged fuel elements to the pool would be displaced upon

impact. Only one percent of released solid fission products would become airborne and released to the environment. This assumption considers that, upon impact, a percentage of the spent nuclear fuel fails, the solid fission products enter the pool, and only finely crushed particulates are splashed out of the pool in the same timeframe that the aircraft hits the water.

- The release to the environment occurs at a constant rate over a 15-minute period.
- Spent nuclear fuel elements remained covered in the water pool.
- The building confinement is assumed to have failed; no filtration by high-efficiency particulate air filters is assumed.

F.6.4.4.4 Fuel Cask Drop

Dry Storage:

Description of Conditions: Mechanical damage due to handling during examination, such as dropping of the spent nuclear fuel cask during transfer, was assessed. The fuel casks are certified to result in no failure for a specific drop height, (free drop from 9 m [30 ft] height onto an unyielding surface), and under no circumstances will the cask be moved above such height during operations within a storage facility. Nevertheless, it was assumed that, upon cask drop, the seals of the cask would fail, releasing the gaseous fission products from the damaged fuel inside the cask to the facility building and the environment. All of the nonvolatile and solid nuclides are assumed to be retained in the fuel or the facility high-efficiency particulate air filters. The resulting airborne release to the environment was evaluated.

Likelihood: The frequency of this scenario is estimated at 10^{-4} per year (DOE, 1995g). This estimate is considered to be an upper bound for this scenario.

Source Term: Conditions used in developing the source term are as follows:

- Only one fuel cask is involved. This is because only one fuel cask is being handled at a time. For the purposes of this analysis, it was assumed that an equivalent of one spent nuclear fuel element inside the cask is damaged, and its gaseous fission products are released inside the cask. This assumption is conservative, since the fuel is secured inside the cask and the cask is not expected to be damaged.
- All (100 percent) of the gaseous fission products and 25 percent of halogens from the damaged fuel element are released to the atmosphere.
- None of the particulate fission products are released to the environment.
- Cs and Ru behave like particulate fission products.
- The release to the environment occurs at a constant rate over a 15-minute period.

Wet Storage:

The source term for a fuel cask drop is similar to that for the fuel element breach scenario in a wet storage facility. The gaseous fission products released inside the cask are vented under water (or in the pool). Since the estimated frequency of this scenario is less than that of the fuel element breach, no specific analysis for this scenario was performed.

F.6.5 Incident-Free Operation Source Terms

This section details the assumptions and the evaluation process used to determine the risk of radiological emissions generated during different activities in incident-free operation of a storage facility. The incident-free operation emissions consist of two parts: transient (i.e., emissions from gaseous release during receipt and unloading of the transportation casks), and steady state (i.e., emissions from spent nuclear fuel in storage). Since only mechanically sound spent nuclear fuel elements are shipped, no radioactive releases are expected during transit. To ensure this, the spent nuclear fuel elements are checked prior to shipment to identify and separate any damaged fuel elements. The damaged fuel elements are then encapsulated and prepared for shipment. In spite of the fact that no spent nuclear fuel elements have ever failed during transit, it was assumed that one percent of the spent nuclear fuel elements will arrive failed and release gaseous fission products (noble gases and halogens) into the cask. Depending on the type of storage facility, the receipt and unloading of the transportation casks could occur in a dry cell or a wet pool. Unloading operations in a dry cell causes all gaseous fission products to be released to the building and eventually to the environment. If the unloading process occurs in a wet pool, a majority of the halogen gases will be absorbed in the water; only 10 percent of halogens will be released to the environment. The building high-efficiency particulate air filters will not be effective for halogens and noble gases. During the unloading process, all spent nuclear fuel elements are checked to ensure that they are mechanically sound. If a damaged fuel element is found, it is encapsulated in a can before it is placed in wet or dry storage. The potential annual radiological releases from failed fuel elements during the unloading process were estimated based on the gaseous inventories of bounding fuels (see Appendix B, Section B.1.4) ad the associated number of fuels expected over the acceptance period. The receipt and unloading process of foreign research reactor spent nuclear fuel from abroad is expected to last 13 years (see Section 2.2.1). It was assumed that failed fuel would release 100 percent of its noble gases and 25 percent of its halogens. This assumption is consistent with that used in the accident analysis.

The steady state emissions from a new wet storage facility are assumed to be similar to those released from the RBOF facility at the Savannah River Site. Although the emissions at the RBOF facility may not be a good representation of the foreign research reactor spent nuclear fuel, RBOF has the most foreign research reactor spent nuclear fuel elements stored in its pool; and as such, was considered to provide the best approximation of the expected release. Based on the emission data from RBOF, the steady-state emissions from a wet storage facility are assumed to be about 2×10^{-7} curies of Cesium-137 per year (DOE, 1995g). This is a conservative assumption. For existing wet storage facilities, the radiation exposure to the MEI and the general public were estimated based on the combined radionuclide atmospheric emissions originating from current conditions of the facilities and that expected from foreign research reactor spent nuclear fuel. At Savannah River Site, the average annual atmospheric emissions from the existing fuels at L-reactor disassembly basin are estimated to be 254 curies of tritium and 6.49×10^{-5} curies of Cesium-137 (Shedrow, 1994b) over Phase 1 of the policy period. The assumption is that the foreign research reactor spent nuclear fuel would be stored temporarily (about 10 years) in the wet pool until a more permanent dry storage facility is built. The annual atmospheric radiological emissions from RBOF and BNFP wet pools are similar to those that are currently released from RBOF and which were used for a new facility. The annual atmospheric radiological emissions from the Idaho National Laboratory's FAST wet storage facility were assumed to be similar to that of a new wet storage facility. This facility has been designed and built according to current codes and regulations.

The steady-state emissions from a dry storage facility are considered to be zero. This is because the fuel will be checked to ensure that it is mechanically sound (i.e., no damage) before it is placed into dry storage, and the dry storage canisters that house the fuel are sealed.

F.6.6 Dose Calculations and Results

F.6.6.1 Source Terms

Tables F-107 and F-108 provide the incident-free operation and accident source terms. The source terms for the annual emissions from the unloading process were calculated based on the assumption that a constant annual rate of fuel mix with bounding radionuclide inventories (as defined in Appendix B, Section B.1) is received over the acceptance period. The fission products in a BR-2 type spent nuclear fuel element were used as the MAR in the accident analysis source term calculations (see Appendix B for more details). Four fuel categories were defined in Appendix B: BR-2, NRU, RHF, and TRIGA. BR-2 fuel type constitutes the majority of the foreign research reactor spent nuclear fuels. In addition, since spent nuclear fuels come in different sizes and lengths, use of the BR-2 spent nuclear fuel in the accident analysis means involvement of a larger number of spent nuclear fuels in each accident. For example, in the source term calculations for an aircraft crash accident involving a transfer cask, it was assumed that the cask would contain 20 BR-2 spent nuclear fuel elements. If the cask contained NRU elements, there would be five elements in the cask; and if it contained RHF elements, there would be only four elements per cask. For the generic wet storage case, the bounding spent nuclear fuel is considered to have been cooled at least 300 days prior to shipment. In the case of dry storage, the fuel has been cooled for at least 3 years.

Table F-107 Annual Emission Releases From Storage Facilities

Isotope	Releases During Unloading in: Dry Cell (Ci)	Releases During Unloading in: Wet Pool (Ci)	Steady State: Dry Storage (Ci)	Steady State: Wet Storage (Ci)
Tritium	39.6	39.6	---	---
Krypton-85	1.14×10^{3}	1.14×10^{3}	---	---
Iodine-129	4.87×10^{-4}	4.87×10^{-5}	---	---
Iodine-131	9.12×10^{-4}	9.12×10^{-5}	---	---
Xenon-131	2.01×10^{-5}	2.01×10^{-5}	---	---
Cesium-137	---	---	0.0	2.20×10^{-7}

Table F-108 Accident Source Terms (Curies)

Isotope	Dry Storage[a]: Fuel Element Breach	Dry Storage[a]: Dropped Fuel Cask	Dry Storage[a]: Aircraft Crash with Fire	Wet Storage: Accident Criticality[b]	Wet Storage: Fuel Element Breach	Wet Storage: Aircraft Crash
Tritium	2.40	2.12	42.4	24.0	2.40	336
Krypton 85	68.6	59.7	1,190	686	68.6	9,610
Iodine 129	3.04×10^{-5}	2.93×10^{-5}	2.34×10^{-3}	3.04×10^{-4}	3.04×10^{-6}	4.26×10^{-4}
Iodine 131	8.88×10^{-8}		0.0	2.20	8.88×10^{-9}	1.24×10^{-6}
Xenon 131m	3.71×10^{-5}		0.0	8.24×10^{-2}	3.71×10^{-5}	5.19×10^{-3}
Strontium 89	1.13×10^{-2}		1.42×10^{-6}	5.67×10^{-3}		0.159
Strontium 90	5.78×10^{-3}		2.73×10^{-2}	2.89×10^{-3}		8.09×10^{-2}
Yttrium 90	5.78×10^{-3}		2.73×10^{-2}	2.89×10^{-3}		8.09×10^{-2}
Yttrium 91	2.03×10^{-2}		8.35×10^{-6}	1.01×10^{-2}		0.284
Zirconium 95	2.97×10^{-2}		3.31×10^{-5}	1.49×10^{-2}		0.416
Niobium 95	6.11×10^{-2}		7.15×10^{-5}	3.06×10^{-2}		0.856
Ruthenium 103	2.47×10^{-3}		1.12×10^{-8}	2.47×10^{-3}		3.46×10^{-2}
Rhodium 103m	2.47×10^{-3}		1.12×10^{-8}	2.47×10^{-3}		3.46×10^{-2}

	Dry Storage[a]			Wet Storage		
Isotope	Fuel Element Breach	Dropped Fuel Cask	Aircraft Crash with Fire	Accident Criticality[b]	Fuel Element Breach	Aircraft Crash
Ruthenium 106	5.97×10^{-3}		6.70×10^{-3}	5.97×10^{-3}		8.36×10^{-2}
Rhodium 106m	5.97×10^{-3}		6.70×10^{-3}	5.97×10^{-3}		8.36×10^{-2}
Tin 123	1.19×10^{-4}		8.30×10^{-6}	5.93×10^{-5}		1.66×10^{-3}
Antimony 125	2.47×10^{-4}		7.10×10^{-4}	1.24×10^{-4}		3.46×10^{-3}
Tellurium 125m	5.89×10^{-5}		1.74×10^{-4}	2.94×10^{-5}		8.24×10^{-4}
Tellurium 127m	2.46×10^{-4}		1.55×10^{-5}	1.23×10^{-4}		3.45×10^{-3}
Tellurium 129m	5.25×10^{-5}		2.95×10^{-11}	2.63×10^{-5}		7.35×10^{-4}
Cesium 134	4.56×10^{-3}		1.10	2.28×10^{-3}		6.38×10^{-2}
Cesium 137	5.72×10^{-3}		2.73	2.86×10^{-3}		8.01×10^{-2}
Cerium 141	1.59×10^{-3}		3.53×10^{-10}	7.97×10^{-4}		2.23×10^{-2}
Cerium 144	8.67×10^{-2}		6.25×10^{-2}	4.33×10^{-2}		1.21
Praseodymium 144	8.67×10^{-2}		6.25×10^{-2}	4.33×10^{-2}		1.21
Promethium 147	1.34×10^{-2}		3.78×10^{-2}	6.71×10^{-3}		0.188
Promethium 148m	2.10×10^{-5}		1.70×10^{-10}	1.05×10^{-5}		2.94×10^{-4}
Europium 154	1.72×10^{-4}		7.30×10^{-4}	8.61×10^{-5}		2.41×10^{-3}
Europium 155	3.61×10^{-5}		1.32×10^{-3}	1.81×10^{-5}		5.06×10^{-4}
Uranium 234	2.50×10^{-10}		1.80×10^{-9}	1.27×10^{-10}		3.55×10^{-9}
Uranium 235	3.80×10^{-9}		1.90×10^{-8}	1.90×10^{-9}		5.37×10^{-8}
Uranium 238	9.50×10^{-11}		4.70×10^{-10}	4.70×10^{-11}		1.33×10^{-9}
Plutonium 238	1.78×10^{-5}		8.75×10^{-5}	8.92×10^{-6}		2.50×10^{-4}
Plutonium 239	5.11×10^{-7}		2.57×10^{-6}	2.56×10^{-7}		7.16×10^{-6}
Plutonium 240	3.33×10^{-7}		1.68×10^{-6}	1.67×10^{-7}		4.67×10^{-6}
Plutonium 241	7.89×10^{-5}		3.56×10^{-4}	3.94×10^{-5}		1.10×10^{-3}
Americium 241	1.10×10^{-7}		1.85×10^{-6}	5.50×10^{-8}		1.54×10^{-6}
Americium 242m	2.92×10^{-10}		1.44×10^{-9}	1.46×10^{-10}		4.08×10^{-9}
Americium 243	1.20×10^{-9}		6.00×10^{-9}	6.01×10^{-10}		1.68×10^{-8}
Curium 244	3.69×10^{-7}		2.25×10^{-7}	1.85×10^{-7}		5.17×10^{-6}
Curium 242	4.86×10^{-7}		6.40×10^{-8}	2.43×10^{-7}		6.81×10^{-6}
Krypton 83m				160		
Krypton 85m				150		
Krypton 87				990		
Krypton 88				650		
Krypton 89				42,000		
Xenon 133m				1.80		
Xenon 133				27.0		
Xenon 135m				2,200		
Xenon 135				360		
Xenon 137				49,000		
Xenon 138				13,000		
Iodine 132				275		
Iodine 133				40.0		
Iodine 134				1,100		
Iodine 135				120		

a *Source terms are those of modular dry vault storage. The dry cask source terms for accident scenarios are the same or smaller than those of modular dry vault storage, therefore, the modular dry vault storage source term values are considered to be bounding values for the impact evaluations.*

b *Particulate source terms (from Strontium 89 to Curium 242) are 1000 times higher, if a facility does not have or has an ineffective, high efficiency particulate air filters. This condition is applicable to the Savannah River Site Wet Storage at RBOF and L-Reactor Disassembly Basin.*

The incident-free operation source terms for chemical separation at the Savannah River Site and the Idaho National Engineering Laboratory were taken from the Interim Management of Nuclear Materials Final EIS (DOE, 1995b) and the Programmatic SNF&INEL Final EIS (DOE, 1995g), respectively. Accident source terms for the chemical separation process were not developed for foreign research reactor spent nuclear fuel. It was considered that the consequences of chemical separation operations-related accidental scenarios are similar to those identified and analyzed in the above documents.

F.6.6.2 Site-Specific Parameters

Several site-specific parameters were required as input to the computer models. The site-specific parameters deal with meteorology, individual and general population food consumption rates, food production locations, and distances and directions of individuals and populations with respect to release locations. The food consumption rates apply only to the MEI and the population dose calculations as indicated in Table F-105. Site-specific food consumption rates consistent with those used in the Programmatic SNF&INEL Final EIS (DOE, 1995g) were utilized. Different contaminated food consumption rates were used at each site because the rate at each site is calculated based on the food production rate within an 80 km (50 mi) radius and the amount of supplemental food (uncontaminated food) that is imported from outside of the 80 km (50 mi) radius. If food production around the site is not sufficient for the population consumption rate, then uncontaminated food is imported. Otherwise, the consumed food is assumed to be contaminated.

F.6.6.3 Results

Tables F-109 through F-116 provide summaries of the consequences, in terms of mrem and/or person-rem, of postulated accident doses to the MEI, NPAI, worker and the public. Except for the worker, where the dose is calculated using the 50th-percentile meteorology, dose calculations were performed for both the 50th- and the 95th-percentile meteorologies using the assumptions and input values discussed above. The accident scenarios and source terms, as described earlier in this appendix, were generically applied to new dry and wet storage facilities. For the existing facilities at each management site, the assumptions and the related source terms were adjusted to conform to the conditions of each facility. Two types of results were provided for the offsite residents (MEI and population). Because protective action guidelines (EPA, 1991) specify mitigative actions to prevent consumption of contaminated food, the dose to offsite residents is reported for all pathways (i.e., external, inhalation, and ingestion) and without the ingestion pathway (i.e., external and inhalation). It should be noted that, as stated earlier, no reduction of exposure to the plume or to contaminated ground surface as a result of early evacuation of offsite populations due to protective action guidelines was accounted for in this analysis.

The analyses were performed for a generic wet and a generic dry storage facility at the Savannah River Site, the Idaho National Engineering Laboratory, the Hanford Site, the Oak Ridge Reservation, and the Nevada Test Site, as well as for site-specific locations (BNFP, L-Reactor Basin area, and RBOF at the Savannah River Site, and FMEF and WNP-4 Spray Pond at the Hanford Site). The consequences of accident scenarios for the IFSF (dry), CPP-749 (dry) and FAST (wet) storage areas at the Idaho National Engineering Laboratory are considered to be equal to those of a generic dry and a generic wet storage facility, respectively. The consequences of accident scenarios for E-MAD at Nevada Test Site are considered to be similar to that of a generic dry storage facility at Nevada Test Site. For the RBOF and the L-Reactor disassembly basin, the criticality accident source terms were adjusted to conform with the conditions assumed in the Basis for Interim Operation reports for these facilities (WSRC, 1995b and 1995c), where no credit was taken for high efficiency particulate air filters after a criticality accident.

Table F-109 Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	***Risk***	***MEI (mrem)[a]***	***NPAI (mrem)***	***Population (person-rem)***	***MEI (mrem)***	***NPAI (mrem)***	***Worker (mrem)***	***Population (person-rem)***
Dry Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event	0.24	0.068	9.2	0.055	0.0043	28	0.62
		Dose/yr	0.038	0.011	1.5	0.0088	0.00069	4.5	0.099
		LCF[b]	1.9×10^{-8}	5.5×10^{-9}	0.00075	4.4×10^{-9}	3.5×10^{-10}	1.8×10^{-6}	0.000050
Dropped Fuel Cask	0.0001	Dose/event	0.018	0.00034	0.55	0.0039	0.000024	0.28	0.011
		Dose/yr	1.8×10^{-6}	3.4×10^{-8}	0.000055	3.9×10^{-7}	2.4×10^{-9}	0.000028	1.1×10^{-6}
		LCF[b]	9.0×10^{-13}	1.7×10^{-14}	2.8×10^{-8}	2.0×10^{-13}	1.2×10^{-15}	1.1×10^{-11}	5.5×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	40	0.29	1300	8.9	0.019	120	87
		Dose/yr	0.000040	2.9×10^{-7}	0.0013	8.9×10^{-6}	1.9×10^{-8}	0.00012	0.000087
		LCF[b]	2.0×10^{-11}	1.5×10^{-13}	6.5×10^{-7}	4.5×10^{-12}	9.5×10^{-15}	4.8×10^{-11}	4.4×10^{-8}
New Wet Storage Accidents - H-Area									
Fuel Assembly Breach	0.16	Dose/event	0.0070	0.00039	0.23	0.0016	0.000027	0.14	0.016
		Dose/yr	0.0011	0.000062	0.037	0.00026	4.3×10^{-6}	0.0022	0.00026
		LCF[b]	5.5×10^{-10}	3.1×10^{-11}	0.000019	1.3×10^{-10}	2.2×10^{-12}	8.8×10^{-10}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	17	9.5	370	4.0	0.69	1600	15
		Dose/yr	0.053	0.030	1.2	0.012	0.0021	5.0	0.047
		LCF[b]	2.7×10^{-8}	1.5×10^{-8}	0.00060	6.0×10^{-9}	1.1×10^{-9}	2.0×10^{-6}	0.000024
Aircraft Crash	1×10^{-6}	Dose/event	4.1	0.98	150	0.92	0.061	400	10
		Dose/yr	4.1×10^{-6}	9.8×10^{-7}	0.00015	9.2 x 10-7	6.1×10^{-8}	0.00040	0.000010
		LCF[b]	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	4.6×10^{-13}	3.1×10^{-14}	1.6 x 10-10	5.0×10^{-9}

Table F-109A Summary of the Accident Analysis Dose Assessments at the Savannah River Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	***Risk***	***MEI (mrem)[a]***	***Population (person-rem)***	***MEI (mrem)***	***Population (person-rem)***
Dry Storage Accidents - H- Area						
Fuel Assembly Breach	0.16	Dose/event	0.053	3.1	0.012	0.0011
		Dose/yr	0.0085	0.050	0.0019	0.00018
		LCF[b]	4.3×10^{-9}	0.000025	9.5×10^{-10}	9.0×10^{-8}
Dropped Fuel Cask	0.0001	Dose/event	0.00024	0.015	0.000055	0.00090
		Dose/yr	2.4×10^{-8}	1.5×10^{-6}	5.5×10^{-9}	9.0×10^{-8}
		LCF	1.2×10^{-14}	7.5×10^{-10}	2.8×10^{-15}	4.5×10^{-11}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	0.91	55	0.20	0.037
		Dose/yr	9.1×10^{-7}	0.000055	2.0×10^{-7}	3.7×10^{-8}
		LCF	4.6×10^{-13}	2.8×10^{-8}	1.0×10^{-13}	1.9×10^{-11}
New Wet Storage Accidents - H-Area						
Fuel Assembly Breach	0.16	Dose/event	0.00027	0.017	0.000062	0.0010
		Dose/yr	0.000043	0.0027	9.9×10^{-6}	0.00016
		LCF	2.2×10^{-11}	1.4×10^{-6}	5.0×10^{-12}	8.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	9.9	240	2.5	5.8
		Dose/yr	0.031	0.74	0.0078	0.018
		LCF	1.6×10^{-8}	0.00037	3.9×10^{-9}	9.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	0.76	46	0.18	3.1
		Dose/yr	7.6×10^{-7}	0.000046	1.8×10^{-7}	3.1×10^{-6}
		LCF	3.8×10^{-13}	2.3×10^{-8}	9.0×10^{-14}	1.6×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-110 Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			50th-Percentile Meteorology			
	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	1.3	0.67	15	0.093	0.062	28	0.83
		Dose/yr	0.21	0.11	2.4	0.015	0.0099	4.5	0.13
		LCF[b]	1.1×10^{-7}	5.5×10^{-8}	0.0012	7.5×10^{-9}	5.0×10^{-9}	1.8×10^{-6}	0.000065
Dropped Fuel Cask	0.0001	Dose/event	0.074	0.0033	0.83	0.0052	0.00032	0.12	0.047
		Dose/yr	7.4×10^{-6}	3.3×10^{-7}	0.000083	5.2×10^{-7}	3.2×10^{-8}	0.000012	4.7×10^{-6}
		LCF	3.7×10^{-12}	1.7×10^{-13}	4.2×10^{-8}	2.6×10^{-13}	1.6×10^{-14}	4.8×10^{-12}	2.4×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	2.9	2000	13	0.27	120	110
		Dose/yr	0.00018	2.9×10^{-6}	0.0020	0.000013	2.7×10^{-7}	0.00012	0.00011
		LCF	9.0×10^{-11}	1.5×10^{-12}	1.0×10^{-6}	6.5×10^{-12}	1.4×10^{-13}	4.8×10^{-11}	5.5×10^{-8}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.0016	0.0036	0.43	0.0028	0.00036	0.14	0.025
		Dose/yr	0.00026	0.00058	0.069	0.00045	0.000058	0.022	0.0040
		LCF	1.3×10^{-10}	2.9×10^{-10}	0.000035	2.3×10^{-10}	2.9×10^{-11}	8.8×10^{-9}	2.0×10^{-6}
Accidental Criticality	0.0031	Dose/event	28	30	140	3.4	12	1800	12
		Dose/yr	0.087	0.093	0.43	0.011	0.037	5.6	0.037
		LCF	4.4×10^{-8}	4.7×10^{-8}	0.00022	5.5×10^{-9}	1.9×10^{-8}	2.2×10^{-6}	0.000019
Aircraft Crash	1×10^{-6}	Dose/event	22	9.8	250	1.6	0.88	400	14
		Dose/yr	0.000022	9.8×10^{-6}	0.00025	1.6×10^{-6}	8.8×10^{-7}	0.00040	0.00014
		LCF	1.1×10^{-11}	4.9×10^{-12}	1.3×10^{-7}	8.0×10^{-13}	4.4×10^{-13}	1.6×10^{-10}	7.0×10^{-8}

Table F-110A Summary of the Accident Analysis Dose Assessments at the Idaho National Engineering Laboratory Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		50th-Percentile Meteorology	
	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.23	2.7	0.017	0.15
		Dose/yr	0.037	0.43	0.0027	0.024
		LCF[b]	1.9×10^{-8}	0.00022	1.4×10^{-9}	0.000012
Dropped Fuel Cask	0.0001	Dose/event	0.0010	0.013	0.000079	0.00076
		Dose/yr	1.0×10^{-7}	1.3×10^{-6}	7.9×10^{-9}	7.6×10^{-8}
		LCF	5.0×10^{-14}	6.5×10^{-10}	4.0×10^{-15}	3.8×10^{-11}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	4.0	0.45	0.29	2.6
		Dose/yr	4.0×10^{-6}	0.000045	2.9×10^{-7}	2.6×10^{-6}
		LCF	2.0×10^{-12}	2.3×10^{-8}	1.5×10^{-13}	1.3×10^{-9}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.0012	0.014	0.000090	0.00085
		Dose/yr	0.00019	0.0022	0.000014	0.00014
		LCF	9.5×10^{-11}	1.1×10^{-6}	7.0×10^{-12}	7.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	17	26	2.6	5.4
		Dose/yr	0.053	0.081	0.0081	0.017
		LCF	2.7×10^{-8}	0.000041	4.1×10^{-9}	8.5×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	3.4	39	0.25	2.1
		Dose/yr	3.4×10^{-6}	0.000039	2.5×10^{-7}	2.1×10^{-6}
		LCF	1.7×10^{-12}	2.0×10^{-8}	1.3×10^{-13}	1.1×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-111 Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***NPAI (mrem)***	***Population (person-rem)***	***MEI (mrem)***	***NPAI (mrem)***	***Worker (mrem)***	***Population (person-rem)***
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	3.0 0.48 2.4×10^{-7}	0.57 0.091 4.6×10^{-8}	42 6.7 0.0034	0.15 0.024 1.2×10^{-8}	0.061 0.0098 4.9×10^{-9}	50 8.0 3.2×10^{-6}	2.0 0.32 0.00016
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.26 0.000026 1.3×10^{-11}	0.0085 8.5×10^{-7} 4.3×10^{-13}	3.0 0.00030 1.5×10^{-7}	0.011 1.1×10^{-6} 5.5×10^{-13}	0.00031 3.1×10^{-8} 1.6×10^{-14}	0.22 0.000022 8.8×10^{-12}	0.15 0.000015 7.5×10^{-9}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.13 0.021 1.1×10^{-8}	0.0033 0.00053 2.7×10^{-10}	1.6 0.26 0.00013	0.0064 0.0010 5.0×10^{-10}	0.00035 0.000056 2.8×10^{-11}	0.25 0.040 1.6×10^{-8}	0.078 0.013 6.5×10^{-6}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	64 0.20 1.0×10^{-7}	14 0.044 2.2×10^{-8}	740 2.3 0.0012	4.8 0.015 7.5×10^{-9}	12 0.037 1.9×10^{-8}	3600 11 4.4×10^{-6}	55 0.17 0.000085
Aircraft Crash[c]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-111A Summary of the Accident Analysis Dose Assessments at the Hanford Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***Population (person-rem)***	***MEI (mrem)***	***Population (person-rem)***
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[b]	0.30 0.048 2.4×10^{-8}	6.5 1.0 0.00050	0.015 0.0024 1.2×10^{-9}	0.31 0.050 0.000025
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0039 3.9×10^{-7} 2.0×10^{-13}	0.029 2.9×10^{-6} 1.5×10^{-9}	0.000071 7.1×10^{-9} 3.6×10^{-15}	0.0015 1.5×10^{-7} 7.5×10^{-11}
Aircraft Crash w/Fire[c]	NA	---	NA	NA	NA	NA
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0016 0.00026 1.3×10^{-10}	0.032 0.0051 2.6×10^{-6}	0.000079 0.000013 6.5×10^{-12}	0.0018 0.00029 1.5×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	7.9 0.025 1.3×10^{-8}	180 0.56 0.00028	2.0 0.0062 3.1×10^{-9}	27 0.084 0.000042
Aircraft Crash[c]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7}/yr.*

Table F-112 Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			50th-Percentile Meteorology			
	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	22	42	55	2.1	9.4	140	8.4
		Dose/yr	3.5	6.7	8.8	0.34	1.5	22	1.3
		LCF[b]	1.8×10^{-6}	3.4×10^{-6}	0.0044	1.7×10^{-7}	7.5×10^{-7}	8.8×10^{-6}	0.00065
Dropped Fuel Cask	0.0001	Dose/event	1.4	0.18	15	0.14	0.042	0.61	2.3
		Dose/yr	0.00014	0.000018	0.0015	0.000014	4.2×10^{-6}	0.000061	0.00023
		LCF	7.0×10^{-11}	9.0×10^{-12}	7.5×10^{-7}	7.0×10^{-12}	2.1×10^{-12}	2.4×10^{-11}	1.2×10^{-7}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	2300	180	2900	220	41	610	440
		Dose/yr	0.0023	0.00018	0.0029	0.00022	0.000041	0.00061	0.00044
		LCF	1.2×10^{-9}	9.0×10^{-11}	1.5×10^{-6}	1.1×10^{-10}	2.1×10^{-11}	2.4×10^{-10}	2.2×10^{-7}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.71	0.20	16	0.068	0.046	0.68	2.5
		Dose/yr	0.11	0.0032	2.6	0.011	0.0074	0.11	0.40
		LCF	5.5×10^{-8}	1.6×10^{-8}	0.0013	5.5×10^{-9}	3.7×10^{-9}	4.4×10^{-8}	0.00020
Accidental Criticality	0.0031	Dose/event	1500	3300	1400	230	910	6800	210
		Dose/yr	4.7	10	4.3	0.71	2.8	21	0.65
		LCF	2.4×10^{-6}	5.0×10^{-6}	0.0022	3.6×10^{-7}	1.4×10^{-6}	8.4×10^{-6}	0.00033
Aircraft Crash	1×10^{-6}	Dose/event	380	600	2900	29	130	1900	120
		Dose/yr	0.00038	0.00060	0.0029	0.000029	0.00013	0.0019	0.00012
		LCF	1.9×10^{-10}	3.0×10^{-10}	1.5×10^{-6}	1.5×10^{-10}	6.5×10^{-11}	7.6×10^{-10}	6.0×10^{-8}

Table F-112A Summary of the Accident Analysis Dose Assessments at the Oak Ridge Reservation Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		50th-Percentile Meteorology	
	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	9.8	29	0.96	4.4
		Dose/yr	1.6	4.6	0.15	0.70
		LCF[b]	8.0×10^{-7}	0.0023	7.5×10^{-8}	0.00035
Dropped Fuel Cask	0.0001	Dose/event	0.038	0.13	0.0039	0.021
		Dose/yr	3.8×10^{-6}	0.000013	3.9×10^{-7}	2.1×10^{-6}
		LCF	1.9×10^{-12}	6.5×10^{-9}	2.0×10^{-13}	1.1×10^{-9}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	500	17	76
		Dose/yr	0.00018	0.00050	0.000017	0.000076
		LCF	9.0×10^{-11}	2.5×10^{-7}	8.5×10^{-12}	3.8×10^{-8}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.042	0.14	0.0043	0.023
		Dose/yr	0.0067	0.022	0.00069	0.0037
		LCF	3.4×10^{-9}	0.000011	3.5×10^{-10}	1.9×10^{-6}
Accidental Criticality	0.0031	Dose/event	1100	1100	180	150
		Dose/yr	3.4	3.4	0.56	0.47
		LCF	1.7×10^{-6}	0.0017	2.8×10^{-7}	0.00024
Aircraft Crash	1×10^{-6}	Dose/event	140	420	13	61
		Dose/yr	0.00014	0.00042	0.000013	0.000061
		LCF	7.0×10^{-11}	2.1×10^{-7}	6.5×10^{-12}	3.1×10^{-8}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-113 Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***NPAI (mrem)***	***Population (person-rem)***	***MEI (mrem)***	***NPAI (mrem)***	***Worker (mrem)***	***Population (person-rem)***
Dry Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	1.7	0.31	1.5	0.052	0.0046	20	0.038
		Dose/yr	0.27	0.050	0.24	0.0083	0.00074	3.2	0.0060
		LCF[b]	1.4×10^{-7}	2.5×10^{-8}	0.00012	4.2×10^{-9}	3.7×10^{-10}	1.3×10^{-6}	3.0×10^{-9}
Dropped Fuel Cask	0.0001	Dose/event	0.11	0.0014	0.40	0.0033	0.000026	0.089	0.010
		Dose/yr	0.000011	1.4×10^{-7}	0.000040	3.3×10^{-7}	2.6×10^{-9}	8.9×10^{-6}	1.0×10^{-6}
		LCF	5.5×10^{-12}	7.0×10^{-14}	2.0×10^{-8}	1.7×10^{-13}	1.3×10^{-15}	3.6×10^{-12}	5.0×10^{-10}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	180	1.2	250	5.6	0.020	87	6.2
		Dose/yr	0.00018	1.2×10^{-6}	0.00025	5.6×10^{-6}	2.0×10^{-8}	0.000087	6.2×10^{-6}
		LCF	9.0×10^{-11}	6.0×10^{-13}	1.3×10^{-7}	2.8×10^{-12}	1.0×10^{-14}	3.5×10^{-11}	3.1×10^{-9}
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event	0.054	0.0016	0.33	0.0017	0.000029	0.10	0.0084
		Dose/yr	0.0086	0.00026	0.053	0.00027	4.6×10^{-6}	0.016	0.0013
		LCF	4.2×10^{-9}	1.3×10^{-10}	0.000026	1.4×10^{-10}	2.3×10^{-12}	6.4×10^{-9}	6.5×10^{-7}
Accidental Criticality	0.0031	Dose/event	88	15	54	6.9	1.1	1300	1.9
		Dose/yr	0.27	0.047	0.17	0.021	0.0034	4.0	0.0059
		LCF	1.4×10^{-7}	2.3×10^{-8}	0.000084	1.1×10^{-8}	1.7×10^{-9}	0.000016	3.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	29	4.2	61	0.92	0.067	290	1.6
		Dose/yr	0.000029	4.2×10^{-6}	0.000061	9.2×10^{-7}	6.7×10^{-8}	0.00029	1.6×10^{-6}
		LCF	1.5×10^{-11}	2.1×10^{-12}	3.1×10^{-8}	4.6×10^{-13}	3.4×10^{-14}	1.2×10^{-10}	8.0×10^{-10}

Table F-113A Summary of the Accident Analysis Dose Assessments at the Nevada Test Site Generic Storage Facilities - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***Population (person-rem)***	***MEI (mrem)***	***Population (person-rem)***
Dry Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.78	0.26	0.024	0.0066
		Dose/yr	0.13	0.042	0.0038	0.0011
		LCF[b]	6.2×10^{-8}	0.000021	1.9×10^{-9}	5.3×10^{-7}
Dropped Fuel Cask	0.0001	Dose/event	0.0031	0.0011	0.00011	0.000033
		Dose/yr	3.1×10^{-7}	1.1×10^{-7}	1.1×10^{-8}	3.3×10^{-9}
		LCF	1.6×10^{-13}	5.5×10^{-11}	5.5×10^{-15}	1.7×10^{-12}
Aircraft Crash w/Fire	1×10^{-6}	Dose/event	13	4.5	0.41	0.12
		Dose/yr	0.000013	4.5×10^{-6}	4.1×10^{-7}	1.2×10^{-7}
		LCF	6.5×10^{-12}	2.3×10^{-9}	2.1×10^{-13}	6.0×10^{-11}
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event	0.0036	0.0013	0.00012	0.000037
		Dose/yr	0.00058	0.00021	0.000019	5.9×10^{-6}
		LCF	2.9×10^{-10}	1.1×10^{-7}	9.5×10^{-12}	3.0×10^{-9}
Accidental Criticality	0.0031	Dose/event	55	5.4	5.8	0.70
		Dose/yr	0.17	0.017	0.018	0.0022
		LCF	8.5×10^{-8}	8.5×10^{-6}	9.0×10^{-9}	1.1×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event	11	3.7	0.35	0.096
		Dose/yr	0.000011	3.7×10^{-6}	3.5×10^{-7}	9.6×10^{-8}
		LCF	5.5×10^{-12}	1.9×10^{-9}	1.8×10^{-13}	4.8×10^{-11}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-114 Summary of the Accident Analysis Dose Assessments at the Barnwell Nuclear Fuels Plant Wet Storage Facility[a] at the Savannah River Site - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)[b]*	*NPAI (mrem)*	*Population (Person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (Person-rem)*
Wet Storage Accidents									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[c]	0.018 0.0056 2.8×10^{-9}	0.00099 0.00016 8.0×10^{-11}	0.028 0.0045 2.3×10^{-6}	0.0055 0.00088 4.4×10^{-10}	0.00027 0.000043 2.2×10^{-10}	0.00080 0.00013 5.2×10^{-11}	0.0033 0.00053 2.7×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	80 0.25 1.3×10^{-7}	75 0.23 1.2×10^{-7}	44 0.14 0.000070	42 0.13 6.5×10^{-8}	45 0.14 7.0×10^{-8}	75 0.23 9.2×10^{-8}	5.6 0.017 8.5×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	92 0.00092 4.6×10^{-10}	31 0.000031 1.6×10^{-11}	23 0.000023 1.2×10^{-8}	11 0.00011 5.5×10^{-11}	3.9 3.9×10^{-6} 2.0×10^{-12}	70 0.000070 2.8×10^{-10}	2.3 2.3×10^{-6} 1.2×10^{-9}

Table F-114A Summary of the Accident Analysis Dose Assessments at the Barnwell Nuclear Fuels Plant Wet Storage Facility[a] at the Savannah River Site - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)[b]*	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Wet Storage Accidents						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[c]	0.00072 0.00012 6.0×10^{-11}	0.0021 0.00034 1.7×10^{-7}	0.00024 0.000038 1.9×10^{-12}	0.00021 0.000034 1.7×10^{-8}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	64 0.20 1.0×10^{-7}	27 0.084 0.000042	37 0.12 6.0×10^{-8}	3.7 0.012 6.0×10^{-6}
Aircraft Crash	1×10^{-6}	Dose/event Dose/yr LCF	17 0.000017 8.5×10^{-12}	6.9 6.9×10^{-6} 3.5×10^{-9}	2.0 2.0×10^{-6} 1.0×10^{-12}	0.68 6.8×10^{-7} 3.4×10^{-10}

[a] *Emissions will be released through an elevated stack for Accidental Criticality and Fuel Assembly Breach accidents.*

[b] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[c] *Point Estimate of Latent Cancer Fatalities event/yr.*

Table F-115 Summary of the Accident Analysis Dose Assessments at the Receiving Basin for Offsite Fuels and L-Reactor Basin Wet Storage Facilities at the Savannah River Site-All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***NPAI (mrem)***	***Population (person-rem)***	***MEI (mrem)***	***NPAI (mrem)***	***Worker (mrem)***	***Population (person-rem)***
Wet Storage Accidents - RBOF									
Fuel Assembly Breach	0.16	Dose/event	0.0070	0.00039	0.23	0.0016	0.000027	0.14	0.016
		Dose/yr	0.0011	0.000062	0.037	0.00026	4.3×10^{-6}	0.0022	0.00026
		LCF[b]	5.5×10^{-10}	3.1×10^{-11}	0.000019	1.3×10^{-10}	2.2×10^{-12}	8.8×10^{-10}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	130	44	4800	30	2.9	16000	310
		Dose/yr	0.40	0.14	14.9	0.093	0.0090	50	0.96
		LCF	2.0×10^{-7}	7.0×10^{-8}	0.0074	4.7×10^{-8}	4.5×10^{-9}	0.000020	0.00048
Aircraft Crash	1×10^{-6}	Dose/event	4.1	0.98	150	0.92	0.061	400	10
		Dose/yr	4.1×10^{-6}	9.8×10^{-7}	0.00015	9.2×10^{-7}	6.1×10^{-8}	0.00040	0.000010
		LCF	2.1×10^{-12}	4.9×10^{-13}	7.5×10^{-8}	4.6×10^{-13}	3.1×10^{-14}	1.6×10^{-10}	5.0×10^{-9}
Wet Storage Accidents- L-Reactor Basin[c]									
Fuel Assembly Breach/	0.16	Dose/event	0.0093	0.00097	0.14	0.0011	0.00015	0.11	0.022
		Dose/yr	0.0015	0.00016	0.022	0.00018	0.00024	0.018	0.0035
		LCF	7.4×10^{-10}	8.0×10^{-11}	0.000011	8.8×10^{-11}	1.2×10^{-11}	7.1×10^{-9}	1.8×10^{-6}
Accidental Criticality	0.0031	Dose/event	170	120	3000	21	21	14000	440
		Dose/yr	0.527	0.37	9.3	0.065	0.065	43	1.4
		LCF	2.6×10^{-7}	1.9×10^{-7}	0.0047	3.3×10^{-8}	3.3×10^{-8}	0.000017	0.00070
Aircraft Crash	1×10^{-6}	Dose/event	4.2	2.6	93	0.60	0.39	70	14
		Dose/yr	4.2×10^{-6}	2.6×10^{-6}	0.000093	6.0×10^{-7}	3.9×10^{-7}	0.000070	0.000014
		LCF	2.1×10^{-12}	1.3×10^{-12}	4.7×10^{-8}	3.0×10^{-13}	2.0×10^{-13}	2.8×10^{-11}	7.0×10^{-9}

Table F-115A Summary of the Accident Analysis Dose Assessments at the Receiving Basin for Offsite Fuels and L-Reactor Basin Wet Storage Facilities at the Savannah River Site-External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	***Risk***	***MEI (mrem)***[a]	***Population (person-rem)***	***MEI (mrem)***	***Population (person-rem)***
Wet Storage Accidents - RBOF						
Fuel Assembly Breach	0.16	Dose/event	0.00027	0.017	0.000062	0.0010
		Dose/yr	0.000043	0.0027	9.9×10^{-6}	0.00016
		LCF[b]	2.2×10^{-11}	1.4×10^{-6}	5.0×10^{-12}	8.0×10^{-8}
Accidental Criticality	0.0031	Dose/event	38	1900	8.8	120
		Dose/yr	0.12	5.9	0.027	0.37
		LCF	5.9×10^{-8}	0.0029	1.4×10^{-8}	0.00019
Aircraft Crash	1×10^{-6}	Dose/event	0.76	46	0.18	3.1
		Dose/yr	7.6×10^{-7}	0.000046	1.8×10^{-7}	3.1×10^{-6}
		LCF	3.8×10^{-13}	2.3×10^{-8}	9.0×10^{-14}	1.6×10^{-9}
Wet Storage Accidents - L-Reactor Basin[c]						
Fuel Assembly Breach	0.16	Dose/event	0.00034	0.010	0.000041	0.0016
		Dose/yr	0.000054	0.0016	6.6×10^{-6}	0.00026
		LCF	2.7×10^{-11}	8.0×10^{-7}	3.3×10^{-12}	1.3×10^{-7}
Accidental Criticality	0.0031	Dose/event	50	1200	6.5	170
		Dose/yr	0.16	3.72	0.020	0.53
		LCF	7.8×10^{-8}	0.0019	1.1×10^{-8}	0.00026
Aircraft Crash	1×10^{-6}	Dose/event	0.77	28	0.11	4.2
		Dose/yr	7.7×10^{-7}	0.000028	1.1×10^{-7}	4.2×10^{-6}
		LCF	3.9×10^{-13}	1.4×10^{-8}	5.5×10^{-14}	2.1×10^{-9}

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event per year.*

Table F-116 Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - All Pathways

			95th-Percentile Meteorology			*50th-Percentile Meteorology*			
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*NPAI (mrem)*	*Population (person-rem)*	*MEI (mrem)*	*NPAI (mrem)*	*Worker (mrem)*	*Population (person-rem)*
Dry Storage Accidents at FMEF[b]									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF[c]	4.7 0.75 3.7×10^{-7}	2.1 0.34 1.7×10^{-7}	46 7.4 0.0037	0.42 0.067 3.4×10^{-8}	0.25 0.040 2.0×10^{-8}	0.99 0.16 6.4×10^{-8}	5.7 0.91 0.00046
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.2 0.00002 8×10^{-12}	0.032 3.2×10^{-6} 1.6×10^{-12}	3.2 0.00032 3.2×10^{-7}	0.017 1.7×10^{-6} 8.5×10^{-13}	0.0017 1.7×10^{-7} 8.5×10^{-14}	0.0049 4.9×10^{-7} 2.5×10^{-13}	0.41 0.000041 2.1×10^{-8}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]									
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.15 0.024 1.2×10^{-8}	0.0033 0.00053 2.7 x 10-10	1.3 0.21 0.00011	0.018 0.0029 1.5×10^{-9}	0.00060 0.000096 4.8×10^{-11}	0.00024 0.000038 1.5×10^{-11}	0.13 0.021 0.000011
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	97 0.3 1.5×10^{-7}	76 0.24 1.2×10^{-7}	620 1.9 0.00096	20 0.062 3.1×10^{-8}	45 0.14 7.0×10^{-8}	120 0.37 1.5×10^{-7}	160 0.50 0.00025
Aircraft Crash[d]	NA	---	NA	NA	NA	NA	NA	NA	NA

Table F-116A Summary of the Accident Analysis Dose Assessments for the Fuel Material Examination Facility Dry Storage and WNP-4 Wet Storage Facilities at the Hanford Site - External and Inhalation Pathways

			95th-Percentile Meteorology		*50th-Percentile Meteorology*	
	Frequency (event/yr)	*Risk*	*MEI (mrem)*[a]	*Population (person-rem)*	*MEI (mrem)*	*Population (person-rem)*
Dry Storage Accidents at FMEF[b]						
Fuel Assembly Breach	0.016	Dose/event Dose/yr LCFc	0.46 0.074 3.7×10^{-8}	6.6 1.1 0.00055	0.041 0.0066 3.3×10^{-9}	0.79 0.12 0.000060
Dropped Fuel Cask	0.0001	Dose/event Dose/yr LCF	0.0028 2.8×10^{-7} 1.4×10^{-13}	0.04 4.0×10^{-6} 2.0×10^{-9}	0.00025 2.5×10^{-8} 1.2×10^{-14}	0.0057 5.7×10^{-7} 2.9×10^{-10}
Aircraft Crash w/Fire[d]	NA	---	NA	NA	NA	NA
Wet Storage Accidents at WNP-4[b]						
Fuel Assembly Breach	0.16	Dose/event Dose/yr LCF	0.0023 0.00037 1.8×10^{-9}	0.032 0.0051 2.6×10^{-6}	0.00028 0.000045 2.2×10^{-11}	0.0034 0.00054 2.7×10^{-7}
Accidental Criticality	0.0031	Dose/event Dose/yr LCF	32 0.099 5.0×10^{-9}	180 0.56 0.00028	12 0.037 1.9×10^{-8}	120 0.37 0.00019
Aircraft Crash[d]	NA	---	NA	NA	NA	NA

NA = Not Applicable

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Emissions will be released through an elevated stack for Fuel Assembly Breach, Dropped Fuel Cask, and Accidental Criticality Accidents.*

[c] *Point Estimate of Latent Cancer Fatalities event/yr.*

[d] *Aircraft Crash accidents are not applicable to the Hanford Site since their frequency of occurrence is less than 10^{-7} event/yr.*

Table F-117 provides a summary of the consequences of radiation exposure to the public and to the MEI from emissions in wet storage (generic and existing), and dry storage (generic and existing).

Table F-117 Normal Release Dose Assessments and Latent Cancer Fatalities at Storage Sites

	MEI Dose (mrem/yr)	*MEI Risk (LCF/yr)*	*Population Dose (person-rem/yr)*	*Population Risk (LCF/yr)*
Savannah River Site				
Receipt/Unloading at:				
RBOF	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
L-Reactor Basin	7.3×10^{-5}	3.7×10^{-11}	4.6×10^{-3}	2.3×10^{-6}
BNFP	6.5×10^{-4}	3.3×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	1.8×10^{-4}	9.0×10^{-11}	8.6×10^{-3}	4.3×10^{-6}
New Wet Storage Facility	1.1×10^{-4}	5.5×10^{-11}	5.7×10^{-3}	2.8×10^{-6}
Storage at:				
RBOF	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
L-Reactor Basin[a]	3.6×10^{-4}	1.8×10^{-10}	2.2×10^{-2}	1.1×10^{-5}
BNFP	7.5×10^{-9}	3.8×10^{-15}	4.8×10^{-8}	2.4×10^{-11}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	1.2×10^{-9}	6.0×10^{-16}	6.2×10^{-8}	3.1×10^{-11}
Idaho National Engineering Laboratory				
Receipt/Unloading at:				
IFSF (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
FAST (wet storage)	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
CPP-749 (dry storage)	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Dry Storage Facility	5.6×10^{-4}	2.8×10^{-10}	4.5×10^{-3}	2.3×10^{-6}
New Wet Storage Facility	3.8×10^{-4}	1.9×10^{-10}	3.1×10^{-3}	1.6×10^{-6}
Storage at:				
IFSF (dry storage)	0	0	0	0
FAST (wet storage)	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
CPP-749 (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	3.8×10^{-9}	1.9×10^{-15}	3.1×10^{-8}	1.6×10^{-11}
Hanford Site				
Receipt/Unloading at:				
FMEF (dry storage)	2.0×10^{-4}	1.0×10^{-10}	1.1×10^{-2}	5.5×10^{-6}
WNP-4 Spray Pond (wet storage)	2.2×10^{-4}	1.1×10^{-10}	5.8×10^{-3}	2.9×10^{-6}
New Dry Storage Facility	2.5×10^{-4}	1.3×10^{-10}	1.5×10^{-2}	7.5×10^{-6}
New Wet Storage Facility	2.0×10^{-4}	1.0×10^{-10}	1.2×10^{-2}	6.0×10^{-6}
Storage at:				
FMEF (dry storage)	0	0	0	0
WNP-4 Spray Pond (wet storage)	5.9×10^{-10}	3.0×10^{-16}	1.6×10^{-8}	8.0×10^{-12}
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	8.8×10^{-10}	4.4×10^{-16}	6.9×10^{-8}	3.5×10^{-11}
Oak Ridge Reservation				
Receipt/Unloading at:				
New Dry Storage Facility	8.9×10^{-2}	4.5×10^{-8}	8.5×10^{-2}	4.3×10^{-5}
New Wet Storage Facility	6.0×10^{-2}	3.0×10^{-8}	6.1×10^{-2}	3.1×10^{-5}
Storage at:				
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.6×10^{-7}	2.3×10^{-13}	5.0×10^{-7}	2.5×10^{-10}

	MEI Dose (mrem/yr)	MEI Risk (LCF/yr)	Population Dose (person-rem/yr)	Population Risk (LCF/yr)
Nevada Test Site				
Receipt/Unloading at:				
E-MAD (dry storage)	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Dry Storage Facility	7.6×10^{-4}	3.8×10^{-10}	9.3×10^{-4}	4.7×10^{-7}
New Wet Storage Facility	5.2×10^{-4}	2.6×10^{-10}	5.2×10^{-4}	2.6×10^{-7}
Storage at:				
E-MAD (dry storage)	0	0	0	0
New Dry Storage Facility	0	0	0	0
New Wet Storage Facility	4.0×10^{-9}	2.0×10^{-15}	4.7×10^{-9}	2.0×10^{-12}

[a] *L-Reactor basin doses are due to existing conditions; the foreign research reactor spent nuclear fuel contribution would be six orders of magnitude smaller*

F.6.7 Accident Scenarios Involving Target Materials

A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident is applicable to the target materials. The frequency of occurrence of an accident involving target materials is estimated to be 3 percent of the 1×10^{-6} per year frequency figure used in the spent nuclear fuel accident analysis. This is because the number of transfer casks that would involve target material is less than 3 percent of that used for 22,700 spent nuclear fuel elements. Therefore, the frequency of this scenario is less than 10^{-7} per year, and is considered to be unforeseeable. Nonetheless, this accident was analyzed and its consequences at potential storage locations were summarized in Table F-118. The frequency of this accident is set conservatively at 10^{-7} per year.

The process by which target materials are prepared for shipment [i.e., drying and canning of the target material solutions, (see Appendix B, Section B.1.5)] releases all gaseous fission products (noble gases and halogens). In addition, the cans in which target materials would be packed do not require any further cutting when they are received in a storage facility. A review of the hypothetical accident scenarios analyzed for spent nuclear fuel indicates that only the aircraft crash with fire accident would be applicable to the target materials. The cans are never cut, and there are no gaseous fission products; therefore, fuel element breach and fuel cask drop scenarios would not be applicable. In addition, should there be an aircraft crash into the wet storage pool where the target material is stored; or, if an accidental criticality in the pool were to occur, the radioactivity releases would be bound by that of the spent nuclear fuel analyzed for these accidents. This is because the amount of radioactive inventory per target material can is very small compared to that in the bounding spent nuclear fuel. In addition, any releases from the target cans would be absorbed in the pool.

Therefore, a scenario involving an aircraft crash into a dry storage facility with ensuing fire was analyzed for the target materials. The scenario assumptions are similar to those described in Section F.6.4.4.3. Because of the size of each can, it was assumed that the transfer cask involved in the accident would contain 40 cans of target materials containing maximum radionuclide inventories, (i.e., 40 cans of 200 grams of ^{235}U per can cooled for at least 3 years). The overall respirable release fraction is assumed to be 5×10^{-3} (Neuhauser and Kanipe, 1993). Table F-119 shows the radioactivity release source terms for this accident.

Table F-118 Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - All Pathways

			95 Percent Meteorology			50 Percent Meteorology			
Site[c]	Frequency (event/yr)	Risk	MEI (mrem)[a]	NPAI (mrem)	Population (person-rem)	MEI (mrem)	NPAI (mrem)	Worker (mrem)	Population (person-rem)
NTS		Dose/event	180	28	120	5.6	0.45	2000	3.0
		Dose/yr	0.000018	2.8×10^{-6}	0.000012	5.6×10^{-7}	4.5×10^{-8}	0.00020	3.0×10^{-7}
	1×10^{-7}	LCF[b]	9.0×10^{-12}	1.4×10^{-12}	6.0×10^{-9}	2.8×10^{-13}	2.3×10^{-14}	8.0×10^{-11}	1.5×10^{-10}
ORR		Dose/event	2400	4000	3700	230	910	14000	560
		Dose/yr	0.00024	0.00040	0.00037	0.000023	0.000091	0.0014	0.000056
	1×10^{-7}	LCF	1.2×10^{-10}	2.0×10^{-10}	1.9×10^{-7}	1.2×10^{-11}	4.6×10^{-11}	5.6×10^{-10}	2.8×10^{-8}
INEL		Dose/event	130	63	1500	9.3	5.7	2700	84
		Dose/yr	0.000013	6.3×10^{-6}	0.00015	9.3×10^{-7}	5.7×10^{-7}	0.00027	8.4×10^{-6}
	1×10^{-7}	LCF	6.5×10^{-12}	6.3×10^{-12}	7.5×10^{-8}	4.7×10^{-13}	2.9×10^{-13}	1.1×10^{-10}	4.2×10^{-9}
SRS		Dose/event	26	6.3	970	5.8	0.41	2700	66
		Dose/yr	2.6×10^{-6}	6.3×10^{-7}	0.000097	5.8×10^{-7}	4.1×10^{-8}	0.00027	6.6×10^{-6}
	1×10^{-7}	LCF	1.3×10^{-12}	3.2×10^{-13}	4.9×10^{-8}	2.9×10^{-13}	2.1×10^{-14}	1.1×10^{-10}	3.3×10^{-9}

Table F-118A Summary of the Accident Analysis Dose Assessments for the Aircraft Crash Accident with Fire Involving Target Material - External and Inhalation Pathways

			95 Percent Meteorology		50 Percent Meteorology	
Site[c]	Frequency (event/yr)	Risk	MEI (mrem)[a]	Population (person-rem)	MEI (mrem)	Population (person-rem)
NTS		Dose/event	64	22	2.1	0.57
		Dose/yr	6.4×10^{-6}	2.2×10^{-6}	2.1×10^{-7}	5.7×10^{-8}
	1×10^{-7}	LCF[b]	3.2×10^{-12}	1.1×10^{-9}	1.1×10^{-13}	2.9×10^{-11}
ORR		Dose/event	870	2500	83	560
		Dose/yr	0.000087	0.00025	8.3×10^{-6}	0.000056
	1×10^{-7}	LCF	4.4×10^{-11}	1.3×10^{-7}	4.2×10^{-12}	2.8×10^{-8}
INEL		Dose/event	20	230	1.4	12
		Dose/yr	2.0×10^{-6}	0.000023	1.4×10^{-7}	1.2×10^{-6}
	1×10^{-7}	LCF	1.0×10^{-12}	1.2×10^{-8}	7.0×10^{-14}	6.0×10^{-10}
SRS		Dose/event	4.4	270	1.0	19
		Dose/yr	4.4×10^{-7}	0.000027	1.0×10^{-7}	1.9×10^{-6}
	1×10^{-7}	LCF	2.2×10^{-13}	1.4×10^{-8}	5.0×10^{-12}	9.5×10^{-10}

NTS = Nevada Test Site; ORR = Oak Ridge Reservation; INEL = Idaho National Engineering Laboratory; SRS = Savannah River Site

[a] *To convert to sieverts from mrem, divide by 100,000; to convert to person-sieverts from person-rem, divide by 100.*

[b] *Point Estimate of Latent Cancer Fatalities event/ yr.*

[c] *Aircraft crash accidents are not applicable to the Hanford Site since their frequency of occurrence is much less than 10^{-7} event/yr.*

Table F-119 Target Materials Aircraft Crash with Fire Accident Source Terms

Isotope	Curies
Strontium-89	2.4×10^{-4}
Strontium-90	3.1×10^{0}
Yttrium-90	3.1×10^{0}
Yttrium-91	1.4×10^{-3}
Zirconium-95	5.2×10^{-3}
Niobium-95	1.1×10^{-2}
Rubidium-103	2.1×10^{-6}
Rubidium-106	7.9×10^{-1}
Ruthenium-103m	2.1×10^{-6}
Tin-123	1.1×10^{-3}
Antimony-125	8.2×10^{-2}
Tellurium-125m	2.0×10^{-2}
Tellurium-127m	2.2×10^{-3}
Tellurium-129m	6.0×10^{-9}
Cesium-134	6.5×10^{-3}
Cesium-137	3.0×10^{-1}
Cerium-141	7.4×10^{-8}
Cerium-144	7.7×10^{0}
Presidium-144	7.7×10^{0}
Promethium-147	6.3×10^{0}
Promethium-148m	2.5×10^{-9}
Europium-154	1.4×10^{-3}
Europium-155	5.2×10^{-2}
Uranium-234	1.4×10^{-7}
Uranium-235	8.3×10^{-5}
Uranium-238	1.5×10^{-6}
Plutonium-238	3.3×10^{-6}
Plutonium-239	6.2×10^{-4}
Plutonium-240	1.4×10^{-5}
Plutonium-241	1.3×10^{-4}
Americium-241	6.9×10^{-7}
Americium-242m	4.4×10^{-12}
Americium-243	3.1×10^{-12}
Curium-242	6.8×10^{-12}
Curium-244	3.2×10^{-12}

F.7 Costs

The cost of implementing the proposed action is analyzed in this section. For the purpose of the cost analysis, the alternatives described in Section 2.1 of the EIS were adjusted to reflect the Record of Decision on the Programmatic SNF&INEL Final EIS (DOE, 1995g) issued in May 1995. According to this Record of Decision, if foreign research reactor spent nuclear fuel is managed in the United States, the aluminum-based portion would be managed at the Savannah River Site and the TRIGA portion would be managed at the Idaho National Engineering Laboratory. The cost analysis also considers the financing arrangements discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS that would affect the cost to the United States. The cost information is presented as follows:

F.7.1 Summary of Cost Information

F.7.2 Costs of Individual Program Components

F.7.1 Summary of Cost Information

This section presents total costs for the proposed policy and implementation alternatives that would impact the costs. The costs are presented in two parts: 1) minimum discounted costs (base case) for the well-defined program components and integration approaches, and 2) "other cost factors" that are likely but sufficiently uncertain that they cannot be directly included in the minimum discounted costs. The costs are shown as net present values in a consistent accounting framework.

Several important factors are used when estimating costs. These factors are as follows:

- *Site- and Implementation-Specific Facilities* - All costs for management in the United States are for facilities that exist or are planned at either the Savannah River Site or the Idaho National Engineering Laboratory. Costs are allocated to the program in proportion to the share of foreign research reactor spent nuclear fuel managed or transferred at each facility. This allocation of capital and operating costs within larger programs results in lower costs to the program than would be the case for the use of facilities dedicated to foreign research reactor spent nuclear fuel.

- *Schedule of Activities* - For all management alternatives (except total management overseas), all spent nuclear fuel is shipped, managed for 40 years, and disposed (either as spent nuclear fuel or as reprocessing waste) on schedules that are appropriate for the selected facilities.

- *Discount Rate* - The base case costs are discounted to 1996 at the rate specified by the Office of Management and Budget for the year ending February 1996. This rate is 4.9 percent real. The base case costs for management outside the United States are discounted at a 3 percent real rate of interest. This rate is estimated to be the long-term real rate of interest that can be expected on a trust fund outside the United States. If the net present value of the costs of the program are received in 1996, a hypothetical trust fund invests the money at the real discount rate so that future expenditures are made out of principal and accrued interest.

- *Net Present Value* - Net present value is a figure-of-merit for decision-making on the basis of life-cycle cost, not a value used for establishing budgets or cash flows. All costs are shown in constant 1996 dollars discounted to 1996. This means that the costs for the duration of the program, expressed as a net present value, are due and payable on January 1, 1996, not in the year the costs are incurred.

- *Timing of Expenses* - All costs are assumed to be incurred on the last day of each year of the 40-year management period. The principal and accrued interest in the trust funds (at the net present value of the program costs) are exactly sufficient to meet the costs as they are incurred.

- *Timing of Payments* - Deferring payments beyond January 1, 1996 increases the payments required (either from reactor operators or the United States Congress) by a factor based on the discount rate and the deferral. Pro-forma full-cost recovery fees are shown for payments made on December 31 of each of the 13 receipt years (1996 through 2008).

- *Inflation and Escalation* - Costs are expressed in constant 1996 dollars in this analysis, so the effects of inflation are eliminated. No costs are escalated in real terms.

- *Ultimate Disposition* - Estimated costs for geologic disposal of intact spent nuclear fuel or waste from chemical separation are included to provide a complete life-cycle cost analysis.

F.7.1.1 Scenarios Analyzed

For the purpose of the cost analysis, six scenarios were analyzed. The scenarios reflect the alternatives that affect cost directly, are consistent with the Record of Decision of the Programmatic SNF&INEL Final EIS (DOE, 1995g) and include the costs for ultimate disposal. The six cost scenarios are:

1. *Management Alternative 1 (Storage)* - Storage of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site with new dry or wet storage facilities; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory at existing wet or dry storage facilities.

2. *Management Alternative 1 (revised to incorporate chemical separation)* - Chemical separation of aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

3. *Management Alternative 1 (revised to incorporate a new technology)* — Implementation of a new treatment and/or packaging technology for aluminum-based foreign research reactor spent nuclear fuel and target material at the Savannah River Site; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

4. *Target Material* - Storage of target material at the Savannah River Site. This scenario provides the cost differential that can be used to assess the cost of managing target material in addition to the foreign research reactor spent nuclear fuel in Management Alternative 1 storage and chemical separation scenarios.

5. *Management Alternative 2* - Management of all foreign research reactor spent nuclear fuel overseas. This scenario reflects a combination of reprocessing and dry storage overseas. Countries with the capability to accept the waste from reprocessing are assumed to have their spent nuclear fuel reprocessed. The rest use dry storage.

6. *Management Alternative 3* - Chemical separation of a portion of the aluminum-based foreign research reactor spent nuclear fuel at the Savannah River Site; reprocessing of the remainder of aluminum-based foreign research reactor spent nuclear fuel overseas; storage of TRIGA foreign research reactor spent nuclear fuel at the Idaho National Engineering Laboratory.

By varying the quantities of material managed in different ways in the United States and overseas, different cost scenarios can be generated. The costs of these variations are bounded by the costs of the scenarios described above. For instance, a management alternative that includes acceptance of target material into the United States would be represented by a combination of Scenarios 1 and 4 or 2 and 4.

The implementation alternatives under Management Alternative 1 related to alternative amounts of foreign research reactor spent nuclear fuel eligible under the policy (Section 2.2.2.1), and alternative policy durations (Section 2.2.2.2), were not considered separately in the cost analysis because they are bounded by the cost scenarios analyzed. These implementation alternatives reduce the amount of foreign research reactor spent nuclear fuel eligible under the policy.

The implementation alternative under Management Alternative 1 related to alternative locations for taking title to the foreign research reactor spent nuclear fuel (Section 2.2.2.4) was not considered because it does not affect the cost analysis.

F.7.1.2 Minimum Program Costs

Table F-120 shows the minimum discounted program costs (base case) for the six scenarios defined above. These costs cover all foreign research reactor spent nuclear fuel shipments, management over 40 years, and geologic disposal. Uncertainties (risks) and escalation are zero. The schedule for activities in Europe under Management Alternative 3 is similar to that in the United States but not exactly the same. Reprocessing takes place over 13 years at Dounreay (the same timespan used for chemical separation at the Savannah River Site) although it could be completed at Dounreay in 9 or 10 years. Dounreay's charges for reprocessing are based on 1996 costs, not costs for 1996 through 2008 averaged over the 13-year period (as was done for the Savannah River Site). Geologic disposal takes place in 2025 through 2030 in Europe and 2030 through 2035 in the United States. Costs are discounted at 3 percent for the portion to be managed overseas and at 4.9 percent for the portion to be managed in the United States.

Table F-120 Minimum Program Costs
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Net Present Value*
1. Management Alternative 1 (Storage)	725/775[a]
2. Management Alternative 1 (revised to incorporate Chemical Separation)	625
3. Management Alternative 1 (revised to incorporate a New Technology)[b]	625-950
4. Target Material	35
5. Management Alternative 2	1,250
6. Management Alternative 3	675

[a] *Dry/Wet new storage facilities*

[b] *Includes target material*

Because of the uncertainties involved with the implementation of the new technology, the cost for Scenario 3 is presented as a range as discussed in Appendix F, Section F.7.2.9. Also, shipping costs in Scenario 3 include the assumption that of the total number of cask shipments, only 38 cask shipments would be accepted at the West Coast.

F.7.1.3 Other Cost Factors

There are four important sources of cost risk (excluding escalation) that are not part of the minimum costs in Table F-120. Table F-121 shows the likely values (risks) for these factors, taking into account the absolute values of the uncertainties and their probability of occurrence. A brief summary of these cost factors follows the table.

The other cost factors summarized in Table F-121 are as follows:

1. *Systems Integration and Logistics Risks* - Significant risks exist in the details of the policy implementation. The implementation of the policy would involve up to 41 foreign countries, up to 13 years of receipts, dozens of foreign ports, up to ten domestic ports, two U.S.

Table F-121 Other Cost Factors
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	Cost Factors: Systems Integration & Logistics	Component Risks	Non-Program Risks	3% Discount Rate	Range
1. Management Alternative 1 (Storage)[a]	100	75	35	175	385
2. Management Alternative (revised to incorporate Chemical Separation)	100	±15	10	125	200-250
3. Management Alternative 1 (revised to incorporate a New Technology)[b,c]	100	75	35	225	435
4. Target Material	5	5	0	25	35
5. Management Alternative 2	100+	±500	1000	250	350-1850
6. Management Alternative 3	100	±10	150	75	315-335

[a] *It is assumed that risks are the same for dry or wet storage options.*

[b] *It is assumed that risk factors are the same as Management Alternative 1 (Storage).*

[c] *Includes target material.*

management sites, and possibly several new facilities. Technical and procedural bottlenecks could arise in many areas.

2. *Component Risks* - Significant risks exist for specific components of the foreign research reactor spent nuclear fuel program, e.g., the adequacy of the characterization of spent nuclear fuel for interim storage, the methods of spent nuclear fuel disposal, the cost allocation at existing and new facilities, and development of new technology.

3. *Non-Program Risks* - Significant risks exist for components of other programs that affect the implementation of the foreign research reactor spent nuclear fuel EIS, e.g., escalating repository costs, adoption of monitored retrievable storage, and differences in facility utilization plans between this EIS and those of other EISs affecting the Savannah River Site and the Idaho National Engineering Laboratory. For Scenario 5, the risks are that no spent nuclear fuel infrastructure exists in more than half of the eligible countries and that no geologic disposal program exists in most of the eligible countries.

4. *Discount Rate Risks* - Significant risks exist that the discount rate required by the Office of Management and Budget for the year ending February, 1996 (4.9 percent real) will be reduced to a more historically representative level (e.g., 3 percent) in some future annual update. The base case costs for management outside the United States are discounted at a 3 percent rate. The use of a high discount rate is particularly risky because 1) revenues are likely to be fixed (in $/kgTM) early in the program while expenses are variable and uncertain, and 2) revenues received from the reactor operators during the 1996 through 2008 shipping period will almost certainly exceed the costs of management activities during that period. Mathematically, the excess revenues are placed in a trust fund that compounds interest at the discount rate. If the discount rate exceeds the rate at which funds actually compound, then outyear program costs (e.g., disposal) could not be met from the principal and accrued interest in the trust fund. A reduction in the discount rate from 4.9 percent to 3.0 percent has a larger impact on the program than any of the technical or systems integration risks.

F.7.1.4 Potential Total Costs

Table F-122 combines the base case costs with the "other cost factors" to provide a realistic expectation of the potential total costs of the program, excluding escalation. The "other cost factors" are divided into technical factors and discount rate-related factors. This table also shows the cumulative percentage effect on the minimum discounted program costs of real escalation at a rate of 1 percent per year over 40 years.

Table F-122 Potential Total Costs
(Net Present Value, Millions of 1996 Dollars in 1996)

Scenario	*Minimum Program Cost*	*Other Cost Factors (Technical)*	*Other Cost Factors (Discount Rate)*	*Potential Total Cost, No Escalation*	*1% Real Escalation, Cumulative*
1. Management Alternative 1 (Storage)	725/775[a]	210	175	≈1,100	+11%
2. Management Alternative 1 (revised to incorporate chemical separation)	625	85-145	125	≈900	+9%
3. Management Alternative 1 (revised to incorporate a new technology)[c]	625-950	210	225	≈1,050-1,400	10%-11%
5. Management Alternative 2	1250	600-1600	250	2,100-3,100	+13%
6. Management Alternative 3[b]	675	225-275	75	≈1,000	+9%

[a] *Dry/Wet new storage facilities.*

[b] *The total cost risk to the United States is less than 1/2 the total cost risk since a large portion of the activities under this alternative would occur overseas.*

[c] *Includes target material.*

Table F-122 shows that the net present value of the potential total costs of implementing the program in the United States, including an estimate of program risks but excluding escalation, range from about $900 for Scenario 2 to $1.4 billion for Scenario 3.

Costs for storing foreign research reactor spent nuclear fuel overseas are highly speculative. In addition, the overseas storage costs are always higher than the more centralized management alternatives because of the extremely high cost of safely and securely managing and disposing of small quantities of spent nuclear fuel in dozens of countries.

The program costs presented in Tables F-120, F-121, and F-122 are in constant 1996 dollars, discounted to 1996. This implies that funds required to cover these costs are received in 1996 and explicitly or implicitly placed in a trust fund. If payments into the trust fund are deferred, then they must be larger than if they had been received on January 1, 1996. For example, if payments are made in 13 equal annual installments every December 31 over the 1996 through 2008 shipping and receiving period, then the constant-dollar payments must increase by 37 percent. A composite of payment schedules, e.g., 13 years for developed country reactor operators and pay-as-you-go (for the United States) for all other costs, including developing country costs, has the effect of increasing the required constant-dollar payments by as much as 25 to 50 percent.

F.7.1.5 Cost to the United States

The cost of the proposed policy to the United States would depend on the type of financing arrangement that DOE adopts in implementing the policy and the discount rate at which revenues from reactor operators accrue interest. Alternative financing arrangements are discussed in Sections 2.2.1.2 and 2.2.2.3 of the EIS. Briefly, the financing arrangements considered are:

1. United States bears the full cost of the program for developing countries and charges a competitive fee to developed countries.

2. United States bears the full cost for all countries (*no* fee).

3. United States charges a *full-cost-recovery* fee to all countries.

4. United States bears the full cost of the program for developing countries and charges a *full-cost-recovery* fee to developed countries.

From a practical standpoint, the U.S. cost under financing arrangement 3 above would be zero. The issue would be whether any foreign countries would participate in the program if full-cost recovery exceeded a competitive fee. The first and fourth arrangements are functionally similar, the U.S. cost resulting from the difference in the *competitive versus the full-cost-recovery fee.* The U.S. cost under the second arrangement (no fee) would be the total program cost as discussed earlier. Any fees established by the United States will take place pursuant to a Federal Register notice after the Record of Decision for this EIS.

Table F-123 shows costs to the United States for the minimum program in each of the cost scenarios analyzed (except target material) under a variety of fee schedules. Adding target material to Scenarios 1, 2, 5, or 6 would increase the cost by 3 to 4 percent. Fees of $2,000/kgTM, $5,000/kgTM, $7,500/kgTM, and $10,000/kgTM, including a pass-through of shipping charges (all expressed in constant 1996 dollars and levelized over 13 years), are used to provide a range of estimates for the cost to the United States. These fees do not imply that reactor operators would pay them for management in Europe or the United States, or that the fee established by the United States will be one of these values. They are used for illustration only and suggest a bounding range, exclusive of technical risk factors, discount rate adjustments, and escalation. The cost to the United States, presented in Table F-123, is the sum of: 1) the cost of managing the foreign research reactor spent nuclear fuel from the developing countries, including shipping, and 2) the difference between the revenues received for management of developed country foreign research reactor spent nuclear fuel and the total program cost of managing developed country foreign research reactor spent nuclear fuel, excluding shipping. Including shipping in the U.S. management costs allows management costs for the United States and the United Kingdom to be presented on a comparable basis.

Table F-123 shows that for minimum discounted program costs and fees charged to developed country reactor operators levelized over 13 years, costs to the United States for management of foreign research reactor spent nuclear fuel (and target material in Scenario 3) could range from several hundred million dollars at a fee of $2,000/kgTM to a profit for fees of $7,500/kgTM to $10,000/kgTM. The cost of managing the spent nuclear fuel from the developing countries (including shipping) adds roughly $100 million more to the cost borne by the United States. Excluding Scenario 5, for which all costs and fees are speculative, the table shows that costs to the United States in Management Alternative 3 are significantly lower than for Management Alternative 1. The savings to the United States exist because the United States bears none of the cost of Spent Nuclear Fuel Management in Europe except the cost of blending down the HEU at Dounreay.

If fees in the $2,000 to $10,000 per kgTM range (levelized $1996 dollars) are established and charged over 13 years, the costs to the United States would be as estimated in Table F-123 (excluding target materials) plus any additional cost factors not incorporated in the minimum program costs. These additional cost factors are: 1) technical risks, 2) discount rate-related risks, and 3) escalation. Table F-122 shows that

Table F-123 Costs to the United States for the Minimum Program Under Various Scenarios and Fee Structures (Millions of 1996 Dollars, Net Present Value of Costs in 1996, Fees Levelized Over 1996-2008 Period)

Scenario[a]	Full-Cost Recovery[b]	Levelized Shipping Fee $/kgTM	Levelized Management Fee (excluding shipping) $/kgTM	Net Present Value For Levelized Fee[c] (Developed Countries Only)				No Fee[d]	Total (excluding shipping)
				$2,000/ kgTM	$5,000/ kgTM	$7,500/ kgTM	$10,000/ kgTM	Developed Countries	
1. Management Alternative 1 (Storage)	100	1,500	6,500	325	100	(75)	(250)	475	575
2. Management Alternative 1 (revised to incorporate Chemical Separation)	90	1,500	5,800	275	50	(125)	(300)	425	525
3. Management Alternative 1 (revised to incorporate a New Technology)	90-110	1,700	5,600-9,200	275-550	50-325	(150)-125	(325)-(50)	425-700	500-800
5. Management Alternative 2[e]	500+							1,250 +	1,750+
6. Management Alternative 3[f]	85	1,500	6,000	225	75	(50)	(175)	300	375

[a] *The total mass (kgTM) of foreign research reactor spent nuclear fuel in the various scenarios is approximately as follows: Aluminum-based plus TRIGA: 115,000 kgTM; from developing countries: 15,000 kgTM; from developed countries: 100,000 kgTM; to Dounreay in Management Alternative 3: 37,000 kgTM. The total mass of target material is approximately 3,400 kgTM aluminum-based equivalent and essentially all from developed countries.*

[b] *Full-cost recovery from developed countries only. The United States bears the costs of the developing countries in these cases.*

[c] *Net present value of costs to the United States for management fees paid in 13 equal annual installments on December 31 of the years 1996 through 2008. Add costs in column labeled "Full-Cost Recovery" to generate total cost to the United States (developed and developing countries).*

[d] *As above, implicitly paid by the taxpayers in 13 equal annual installments (to maintain consistency with the payment period of the reactor operators), excluding shipping. The net present value of shipping in Scenarios 1 [Management Alternative 1 (Storage)] and 2 [Management Alternative 1 (revised to incorporate chemical separation)] is $140 Million. The net present value of shipping to the United States only in Scenario 6 is $90 Million. The net present value of shipping in Scenario 3 [Management Alternative 1 (revised to incorporate a new technology)] is $160 Million.*

[e] *There is no defined basis for the charges to the United States for non-U.S. management. Costs to the United States under Management Alternative 2 assume that the United States absorbs the cost to construct and operate independent foreign research reactor spent nuclear fuel storage installations (including all supporting safety, security, transport, health physics, etc. infrastructure) for the 22 countries with no commercial nuclear power programs and that the United States partially subsidizes the other countries, depending on developmental status, commercial nuclear power infrastructure, and other factors.*

[f] *U.S. component of Management Alternative 3 only. Revenues paid to the United States exclude shipping charges. Costs to the United States for management in Europe consist only of the charge to blend down the HEU to LEU ($20 million). European reactor operators using Dounreay are assumed to bear all other costs.*

technical risks could add roughly $100 to $200 million to the costs borne by the United States. Discount rate-related risks are of a similar size. Escalation risks are more uncertain but could be in the same range.

F.7.2 Costs of Individual Program Components

This section provides details on program costs for each of the scenarios outlined in section F.7.1.

F.7.2.1 Programmatic Cost Assumptions

Table F-124 shows programmatic assumptions about costs and the basis for the cost calculations.

Table F-124 Programmatic Assumptions and Bases

Variable	*Assumption*	*Basis*
Year Dollars	1996	Standardized to first year of program.
Discount Rate for Management in the United States	4.9 percent real	Required by Office of Management and Budget for programs beginning between February 1995 and February 1996.
Discount Rate for Management in Europe	3.0 percent real	Representative of long-run average in larger Western European economies.
Rounding of Totals	$25 million	Highlights differences between programs that typically differ by $100 million. No implication of precision.
Component Contingencies	Included in base costs	Standard costing assumption
Program Risks	Not included in base costs	Logistical complexity of program could add 10-15 percent to total costs.
Uncertainties	Not included in base costs	
Risk-adjustment	Not included in base costs	
Escalation	Not included in base costs	
Costs incurred over what period	40 years (1996 to 2035) in United States 35 years (1996 to 2020) in United Kingdom	Maximum length of interim storage
Repository Shipping	2030 to 2035 in United States 2025 to 2030 in United Kingdom	Storage maximum in United States and United Kingdom
Qualification of fuel types for disposal	$10M per type allocated to the program, 5 types in program	Idaho National Engineering Laboratory estimate. The foreign research reactor spent nuclear fuel program is estimated to be responsible for three types of aluminum-based spent nuclear fuel and two types of TRIGA spent nuclear fuel. The types are related to the repository program characteristics.

F.7.2.2 Individual Program Components

The proposed foreign research reactor spent nuclear fuel program consists of many components. Table F-125 outlines the components of the five cost analysis scenarios described in Section F.7.1. Detailed discussions of the individual program components follow the table.

F.7.2.3 Logistics and Program Management

Under Management Alternative 1, the United States would undertake a program where the maximum requirements begin with the shipping of an estimated 837 casks of foreign research reactor spent nuclear

Table F-125 Applicability of Specific Cost Components to the Cost Analysis Scenarios

Component	*Appendix Section*	*Management Alternative 1 (Storage)*	*Management Alternative 1 (revised to incorporate Chemical Separation)*	*Management Alternative 1 (revised to incorporate a New Technology)*	*Management Alternative 2*	*Management Alternative 3*	*Target Material*
Programmatic Assumptions	F.7.2.1	x	x		x	x	x
Logistics and Program Management	F.7.2.3	x	x		x	x	x
Shipping Spent Nuclear Fuel to the United States	F.7.2.4	x	x			x	x
Shipping Spent Nuclear Fuel to the United Kingdom	F.7.2.5				x	x	
Interim Storage at the Savannah River Site	F.7.2.6	x					x
Interim Storage at the Idaho National Engineering Laboratory	F.7.2.7	x	x			x	
Chemical Separation at the Savannah River Site	F.7.2.8		x			x	
New Technology	F.7.2.9			x			x
Reprocessing in the United Kingdom	F.7.2.10				x	x	
High-Level Waste Vitrification and Separation Waste Storage	F.7.2.11		x		x	x	
Disposal of Spent Nuclear Fuel	F.7.2.12	x			x		x
Disposal of Vitrified High-Level Waste	F.7.2.13		x		x	x	
Storage or Reprocessing Overseas	F.7.2.14				x	x	

fuel and 140 casks of target material (977 casks in total) from dozens of ports in 41 countries to as many as 10 ports in the United States and one or more border crossings from Canada. Consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), aluminum-based foreign research reactor spent nuclear fuel would be delivered to the Savannah River Site and TRIGA foreign research reactor spent nuclear fuel would be delivered to the Idaho National Engineering Laboratory. Approximately 815 casks (including target material) would be shipped to the Savannah River Site and 162 to the Idaho National Engineering Laboratory.

Once the foreign research reactor spent nuclear fuel was in transit to the United States and especially once title had been transferred to the United States, numerous regulations covering safety, health, and environmental compliance would take effect. It is estimated that the direct cost of coordinating shipping, ensuring regulatory compliance, providing program documentation, conducting inspections in the United States and overseas, and providing overall program logistical support is about $5 million per year during the active shipping period (exclusive of shipping costs). This cost would be lower if material is shipped to the United Kingdom (Management Alternative 3). For Management Alternative 1, the discounted cost over the 13-year receipt period in the United States would be approximately $50 million. Costs for logistics and program management during the non-receiving period (years 14 through 40) are assumed to be modest and are accounted for as part of the management costs at the U.S. site.

F.7.2.4 Shipping to the United States

Shipping the foreign research reactor spent nuclear fuel and target material to the United States requires an estimated 977 cask shipments. Of this, 837 cask shipments would contain spent nuclear fuel and 140 cask shipments would contain target material.[4] The shipping period would be thirteen years, beginning in 1996. Discounted total shipping costs to and from the United States are estimated at about $140 million for the spent nuclear fuel and about 10 percent more if target material is included.[5] Under Management Alternative 3, where approximately one-third of the spent nuclear fuel casks are shipped to the United Kingdom, costs for shipping the remaining casks to the United States are about $90 million. Costs include cask rental, inland freight by truck in the United States and overseas, ocean transport to and from the United States (except for shipments from Canada, which would go by truck), port handling, security, insurance, administration, and contingencies. Logistics and program management is described in Section F.7.2.3.

The technical requirements and costs associated with shipping differ depending on the point of origin of the spent nuclear fuel. Costs are estimated separately for seven countries and/or regions of the world: Europe, Australia, Japan, Asia (excluding Australia and Japan), Canada, Other Atlantic, and Other Pacific. This section discusses technical issues associated with shipping the foreign research reactor spent nuclear fuel and target material to the United States from each region.

- **Europe** — European regulations for inland freight and ocean freight shipments of spent nuclear fuel have become very strict in recent years and are virtually certain to become more stringent. Requirements for permits, cross-border shipping, consolidation for ocean shipping, and other factors have driven the cost per cask in an unconsolidated movement to far more than that for inland freight in the United States. Considering the European Community's vessel requirements for the spent nuclear fuel shipped in 1995 under the Urgent Relief Environmental Assessment (DOE, 1994i), it is prudent for costing purposes to assume that shipment by chartered vessel rather than regularly scheduled commercial vessel would be required. (Shipment by purpose-built vessel is not likely to be required.) The cost of chartering a ship capable of carrying spent nuclear fuel casks from Europe to the United States' East Coast and handling the casks at the port and on-board is approximately $400,000. This cost can be spread over a maximum of 6 to 8 casks per vessel. For costing purposes, the EIS assumes 6 casks per vessel and two European ports-of-call. European nations account for an estimated 505 casks, 393 containing aluminum-based spent nuclear fuel, 14 containing target material, and 98 containing TRIGA spent nuclear fuel.

- **Australia** — Australia owns a single, large spent nuclear fuel transportation cask and, thus, does not generate a cask rental charge as part of the spent nuclear fuel program. (A charge for a typical transportation cask is assessed, however, to show true costs to undertake the program.) Australia is unlikely to require chartered shipping. Inland freight charges are moderate. Australia would account for 9 casks, all of which would contain aluminum-based foreign research reactor spent nuclear fuel. For cost analysis, the

---

4 *The number of casks required for target material could be reduced by more than half if the material was converted to an oxide form prior to shipping. The estimate of 140 casks is based on a conservative estimate of shipping the material as a calcine.*

5 *Shipping target material increases costs much less than proportionately because most of the target material is in Canada.*

Australian cask is assumed to be shipped as part of larger shipments from Asia. These shipments would carry 6 casks per vessel and call on three ports per transit to the United States.

- **Japan** — The Japan Atomic Energy Research Institute owns two casks that would be used for spent nuclear fuel accepted by the United States. Because the Japan Atomic Energy Research Institute is near the port of export for Japan, inland freight charges would be negligible. Japan would likely require chartered vessels (at least as far as Europe for shipments of spent nuclear fuel to the United States via Europe). Shipment by chartered vessel would be approximately $450,000, or $225,000 per cask. It is estimated that Japan would ship approximately 110 casks to the United States (99 aluminum-based and 11 TRIGA). Japan could choose to acquire more casks to reduce its cost per ocean transit. As with Australia, a cask charge is assigned to show true program costs.

- **Asia** (excluding Australia and Japan) — Asian nations (excluding Japan) would be expected to have relatively low inland freight costs. It is unclear if Asian nations would require chartered vessels. Asian nations (excluding Australia and Japan) account for an estimated 62 casks (23 containing aluminum-based spent nuclear fuel, 1 containing target material, and 38 containing TRIGA spent nuclear fuel).

- **Canada** — For cost analysis, all Canadian shipments (approximately 116 casks of aluminum-based spent nuclear fuel and 125 casks of target material) are assumed to come by truck to the Savannah River Site. Cask rental and inland freight charges reflect the shipping times and distances for long overland routes. Shipping by rail is also feasible.

- **Other Atlantic** — All other nations nearer the Atlantic Ocean than the Pacific Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Shipments from Mexico would come by sea, since the Mexican spent nuclear fuel is located in the southern part of the country. The Other Atlantic nations are not likely to require chartered vessels. Other Atlantic nations account for 38 casks, 23 of which would contain aluminum-based spent nuclear fuel and 15 of which would contain TRIGA spent nuclear fuel.

- **Other Pacific** — All other nations nearer the Pacific Ocean than the Atlantic Ocean are assumed to have characteristics similar to those of Asia (excluding Australia and Japan) but lower ocean shipping costs because of greater proximity to the United States. Because the Other Pacific countries are on the western coast of South America (which is significantly closer to the southeastern United States than the northwestern United States) and because all the spent nuclear fuel from these countries is aluminum-based, the EIS assumes that all shipments from Other Pacific countries will go by sea to an East Coast port via the Panama Canal. The Other Pacific nations would not be likely to require chartered vessels. Other Pacific nations account for 12 casks, all of which would contain aluminum-based spent nuclear fuel.

Table F-126 summarizes the cost of shipping a single spent nuclear fuel cask from various parts of the world to the United States in the configuration considered most likely by this EIS. The base case assumes the use of charter ships. The discounted cost of overseas shipping to the United States (including overland shipping from Canada and including target material) is shown in the table as $158 million (summing the bottom row). Of the 977 shipments, 827 originate either in Canada or in ports nearer the U.S. East Coast

Table F-126 Representative Shipping Costs to/from the United States for a Spent Nuclear Fuel Cask (Thousands of 1996 Dollars per Cask and Millions of 1996 Dollars for the Program, including Target Material)

Activity/Cost	*Europe*	*Australia*	*Japan*	*Other Asia*	*Canada*	*Other Atlantic*	*Other Pacific*
Charter	Y	Y	Y	Y	N/A	Y	Y
U.S. Coast	East	West	West	West	N/A	East	East
Charter Cost $k	400	550	550	500	N/A	300	300
Casks/Charter	6	See Other Asia	6	6	N/A	6	6
Ports-of-Call	2	See Other Asia	1	3	N/A	3	3
Total Rental Charges, $k/Cask	51	48	42	66	21	60	66
Inland Freight, Country, Site, and Overland Route Weighted, $k/Cask	37	38	41	30	25	26	38
Insurance, Security, Administration, Cask Return, $k/Cask	51	49	58	70	36	49	49
$k/Cask, Excluding Contingency	224	253	239	246	86	232	246
Number of Casks (Aluminum)	393	9	99	23	116	23	12
Number of Casks (TRIGA)	98	0	11	38	0	15	0
Number of Casks (Target Material)	14	0	0	1	125	0	0
Number of Casks (Total)	505	9	110	62	241	38	12
of which, from Developing Countries	72	0	0	53	0	38	12
Total Cost, including 15% Contingency, $M	130	2	30	18	24	10	3
Discounted Cost ($M)	95	2	22	13	17	7	2

(including 12 cask shipments from the West Coast of South America). The remaining 150 cask shipments originate in ports nearer the U.S. West Coast. Assuming shipments to the nearest U.S. coast, regardless of the type of spent nuclear fuel, an estimated 113 shipments of TRIGA spent nuclear fuel received at East Coast ports and an estimated 132 shipments of aluminum-based spent nuclear fuel received at West Coast ports would be shipped overland to the appropriate management site. Key issues in analyzing shipping costs follow the table.

- **Use of Chartered Ships** - Chartered vessels are used for conservatism in costing. The cost increase from using charters rather than regularly scheduled commercial vessels (for those countries likely to permit shipping by regularly scheduled commercial liners) is approximately $10 million. Europe and Japan would require charters in any case. Canada would transport by land. These three regions/countries account for more than 80 percent of the proposed shipments. No additional port-related costs are assigned if military ports are used.

- **Casks Per Ocean Shipment** - The costs in Table F-126 assume that European and Japanese shipments are made by a chartered vessel carrying six casks consolidated at two ports in Europe and one in Japan. For Japan, this charter loading implies the acquisition of more casks or the use of commercial casks. The nine casks (total) from Australia are assumed to be part of larger shipments from Asia. Even though Japan and Australia would most likely use their own casks, a cask rental charge is shown to reflect true program costs regardless of where the costs are borne and to reflect the likely requirement for new casks by Japan. Reducing the number of casks per shipment from Japan increases shipping costs by $20 million.

Shipments from the rest of the world (excluding Europe and Canada) are assumed by charter at the rate of 6 casks per vessel and 3 ports-of-call (i.e., two casks per country). Adding ports-of-call increases costs in transit (by about $20,000 per port-of-call and $20,000 per day in transit between ports) but saves money on balance by increasing the number of casks on the ship. Reducing the shipments from Asia (excluding Australia, Japan) and the Other Atlantic and Other Pacific countries to 2 casks and 1 port-of-call would increase program costs by $12 million.

- **Shipping to Distant Coasts and Sites** -- The cost of shipping the foreign research reactor spent nuclear fuel depends on which ports were selected and from where they would be accepting the shipments. The dynamics of the program are that roughly 75 percent of the foreign research reactor spent nuclear fuel is aluminum-based (and therefore would be destined for the Savannah River Site, on the United States' East Coast) and roughly 75 percent of the foreign research reactor spent nuclear fuel (excluding Canadian spent nuclear fuel) is in countries on the Atlantic side of the United States. While the 75 percent aluminum-based spent nuclear fuel and the 75 percent Atlantic spent nuclear fuel are not identical, there is sufficient overlap to create a situation where shipping all the spent nuclear fuel directly to a United States East Coast port and then distributing the TRIGA spent nuclear fuel to the Idaho National Engineering Laboratory by land would be only about 5 percent ($8 million) more expensive than shipping the spent nuclear fuel to the nearest port and then overland to the appropriate site. The cost of overland shipping by truck from an eastern port to the Idaho National Engineering Laboratory for a shipment that would logically arrive at an eastern port is less than the cost of ocean shipping to a western port to minimize the overland transit by truck.

- **Receipt Rates at the Savannah River Site and the Idaho National Engineering Laboratory** -- To accept all the foreign research reactor spent nuclear fuel within the proposed 13-year period requires, on average, cask receipts of almost six casks per month (seven per month if target material is included). Splitting the spent nuclear fuel by fuel type, consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), implies receipt of 4 to 5 casks per month of aluminum-based spent nuclear fuel at the Savannah River Site and about one cask per month of TRIGA spent nuclear fuel at the Idaho National Engineering Laboratory.[6] About 1 cask per month of target material would also be received at the Savannah River Site.

- **Cask Rental Charges** -- Truck casks rent for approximately $1,500 per day on long-term lease. Shorter-term rentals are appreciably more expensive (EG&G, 1994b). Table F-126 incorporates the $1,500 per day rate for a long-term lease. The use of the smaller truck casks (compared to rail casks) permits savings in ocean shipping, short overland transport (although this could change in response to high charter costs), and security. The cost to acquire a new truck cask has been increasing steadily and is now approaching $2 million. The time from ordering to delivery exceeds 1 year. Because of the limited market for casks and the risk of constructing a cask for which there is no long-term demand, potential cask owners and lessors would place a high fixed charge rate on an investment in new casks for the foreign research reactor program. For a 20-year operating life, the fixed

---

6 *The weighted-average number of spent nuclear fuel elements per cask is estimated to be slightly more than 27. The sites are limited by cask receipt rates, not elements per cask. Some casks would have as many as 120 elements. Others would have one element. Most would have about 27 to 30 elements.*

charge rate would be at least 30 percent. For a fixed charge rate of 30 percent, a $2 million cask must rent for $600,000 per year, or approximately $1,650 per day on a yearly lease.

- **Cask Shipment and Rental Periods** -- The average time required to complete a round-trip shipment depends on the area of cask origin, the number of casks shipped at one time, the number of ports-of-call made enroute to the United States, inland shipping in the United States, and turnaround time at the sites. Excluding Canada, round-trip cask shipment periods range from an average of less than 40 days for a cask from the Atlantic coast of South America to the southeast coast of the United States (with an ultimate destination of the Savannah River Site) to more than 60 days for a cask from Australia to the same ultimate destination (either via a Pacific port and an overland transit to the Savannah River Site or via a passage through the Panama Canal to an Atlantic Port).

The base costs cover two ocean transits, port handling in two countries, shipment to and from the cask lessor, and overland transport from the ports to and from the sites and reactor facilities. Cask handling at the sites is estimated separately.

- **Contingencies** -- Over the past few years, the cost of almost all phases of international spent nuclear fuel shipping has risen sharply. Also, European regulations regarding ocean shipping of nuclear cargoes have tightened dramatically. While these costs are built into the values in Table F-126, potentially large additional contingencies are not. These contingencies include escalating cask lease rates; partially filled casks; higher inland freight charges in the United States; dedicated rail shipping in the United States; consolidation limitations in Asia, South America, or Africa; and additional security. On the other hand, the single largest contingency -- the use of charter ships -- has been added to the base case. Consideration of the magnitude of the contingencies suggests a contingency factor of about 15 percent. This factor applies to the shipping component of the program only, not the impacts on the program logistics or integration from delays in shipping, barriers erected by the States, etc. These program-level impacts are discussed separately in Section F.7.4.

F.7.2.5 Shipping to the United Kingdom

Shipping to the United Kingdom is less expensive than shipping to the United States. Cost estimates provided by the United Kingdom Atomic Energy Authority for this EIS are about $30,000 per cask from Europe (Scullion, 1995). This compares to more than $200,000 per cask estimated for shipments from Europe to the United States. The estimates for shipping to the United Kingdom reflect the large savings from the very short ocean transit from continental Europe (and thus the vessel charter cost), the ocean transit and site turn-around periods (and thus the cask rental time), the inland freight charges for shipping a short distance in the United Kingdom, and the reduced administrative, insurance, and security costs for the shorter activity.

It is possible that the estimated cost for shipments to the United Kingdom is understated in comparison to the U.S. costs for at least two reasons. First, no detailed analysis of the cost components similar to that in Table F-126 was conducted and thus some costs, especially indirect costs, such as administration, may have been omitted. Second, costs for shipments to the United States have increased sharply in recent years. Costs for recent shipments to the United States were higher than anticipated and may not be reflected in the estimated costs to ship from Europe to the United Kingdom.

F.7.2.6 Storage at the Savannah River Site

Consistent with the Record of Decision for the Programmatic SNF&INEL Final EIS (DOE, 1995g), approximately 17,800 aluminum-based spent nuclear fuel elements could be received and managed at the Savannah River Site. These elements would be stored or chemically separated. Under Implementation Alternative 1c to Management Alternative 1, target material equal to about 600 aluminum-based elements could also be received and stored at the Savannah River Site. The cost to receive and store the target material is proportional to the ratio of target material (expressed in element-equivalents, e.g., cans) to spent nuclear fuel elements (i.e., about 3.4 percent). Costs in this section refer to the basic implementation of Management Alternative 1 (17,800 spent nuclear fuel elements and no targets).

Storage at the Savannah River Site would consist of two phases: Phase-1 storage in existing facilities and Phase-2 storage in new facilities. Logistically, the base case for Management Alternative 1 (storage) is as follows:

- At the start of the implementation period, aluminum-based spent nuclear fuel would be shipped to the Savannah River Site and wet-stored in RBOF and the L-Reactor disassembly basin.

- At about the same time, construction would begin on a staging and characterization facility and an interim dry or wet storage facility at the Savannah River Site. The staging facility would be designed to receive and transfer all the foreign research reactor spent nuclear fuel (and other nuclear materials, including domestic research reactor spent nuclear fuels). The dry or wet storage facility would be designed to store the spent nuclear fuel (and possibly target material) until the spent nuclear fuel and target material were prepared for shipment to the repository. The new facilities would be commissioned in 2003, accept off-site receipts of foreign research reactor spent nuclear fuel through 2008, and on-site transfers (of all aluminum-based materials, not just foreign research reactor spent nuclear fuel) from the RBOF and the L-Reactor disassembly basin through about 2008 or 2009. If commissioning of the new storage facility is delayed to 2005, transfers from existing basins would continue through about 2010.

- At some point in the 2015 to 2035 time period, the stored spent nuclear fuel would be prepared for repository disposal in as-yet unspecified repository-qualified canisters. Cost estimates are based on a repository packaging and shipping period of 2030 to 2035.

Table F-127 shows the annual costs for storage of 17,800 foreign research reactor spent nuclear fuel elements at the Savannah River Site during Phase 1 and Phase 2, where Phase 2 storage is dry (WSRC, 1995c). Receiving and storing target material would add $20 million (discounted) to expenditures at the Savannah River Site and $35 million (discounted) to the total costs. The key assumptions used to generate the costs in Table F-127 are discussed below.

- Annual operating costs for round-the-clock operations at RBOF and L-Reactor disassembly basin are allocated to the foreign research reactor spent nuclear fuel program in proportion to the share of foreign research reactor spent nuclear fuel mass transferred to or from the basins relative to total cask transfers at RBOF, and L-Reactor disassembly basin in each year until all of the foreign research reactor spent nuclear fuel has been transferred to dry storage (about 2008 or 2009). Unit costs are assumed fixed in each year. Thus, allocable costs scale in proportion to the amount of foreign research reactor material received at the basins.

Table F-127 Storage of Aluminum-Based Spent Nuclear Fuel at the Savannah River Site, Including Phase 2 Dry Storage (Millions of 1996 Dollars)

Year	*Basin Costs*	*Capital Costs-Staging*	*Operating Costs-Staging*	*Capital Costs-Storage Facility*	*Capital Costs-Storage Canisters*	*Operating Costs-Storage Site*	*Decontamination & Decommissioning*
1996	16	1					
1997	20	5					
1998	23	5					
1999	24	6		1			
2000	21	18		1			
2001	15	22		2			
2002	15	24		2			
2003	15	5	3	1	9	3	
2004	15		3	1	9	3	
2005	19		3	1	9	3	
2006	16		3	1	9	3	
2007	16		3	1	9	3	
2008	16		3	1	9	3	
2009	6	1	3	1		3	
2010		1	1			3	
2011		4	1			2	
2012		4	1			2	
2013		3	1			2	
2014		2	1			2	
2015-2029			1/yr.			2/yr.	
2030-2034			3/yr.			3/yr.	
2035			3			3	9
Total Costs (Undiscounted)	238	103	65	14	55	77	9
NPV	174	74	23	9	34	28	1

- A staging facility would be constructed for operation in 2003 (although it could be deferred until 2005). The primary functions of the staging facility would be to: 1) accept on-site transfers and off-site receipts, 2) characterize the spent nuclear fuel, 3) transfer the spent nuclear fuel from the received casks to interim dry or wet storage, and 4) transfer the spent nuclear fuel from interim dry or wet storage to repository-qualified canisters. If processing of the spent nuclear fuel prior to disposal were required, e.g., melting and diluting the material, additional facilities would be added to the staging facility.

  Excluding a melt-and-dilute facility, the staging facility is estimated to have a discounted cost of $150 million. Because the foreign research reactor spent nuclear fuel program would share the facility with other programs, costs are allocated to the foreign research reactor spent nuclear fuel program in proportion to the average of: 1) foreign research reactor spent nuclear fuel to other spent nuclear fuel received and staged to dry or wet storage over the period of facility operations (2003 through 2035), and 2) foreign research reactor spent nuclear fuel to other spent nuclear fuel staged from storage to repository casks over the period of facility operations. Using this approach, about 43 percent of the capital and operating costs of the facility would be allocable to the foreign research reactor spent nuclear fuel program. Foreign research reactor spent nuclear fuel represents about 57 percent of the total aluminum-based spent nuclear fuel to be managed at the Savannah

River Site over the 1996 through 2035 period (by MTR-equivalents) but only 28 percent of the total aluminum-based spent nuclear fuel received initially at the new staging and characterization facility. The unweighted average of these two percentages is 43 percent. Because most of the foreign research reactor spent nuclear fuel arrives prior to the operation of the new staging and characterization facility, the foreign research reactor spent nuclear fuel bears a disproportionately high share of the operating costs of RBOF and L-Reactor disassembly basin and a disproportionately low share of the capital and operating costs of the new staging and characterization facility.

- A new dry or wet storage facility would be constructed for operation in 2003. A dry storage facility would consist of a pad, fence, canisters, and storage overpacks. It would operate through 2035. The canisters used at the dry facility would not necessarily be qualified for repository disposal. During the storage phase, the canisters would be loaded with approximately 228 elements apiece.[7]

For cost analysis, the spent nuclear fuel is assumed to be taken out of storage over the period 2030 through 2035 for transfer to repository-qualified canisters. The actual timing of the transfers could be earlier but cannot be specified at present. The undiscounted cost of the facility per canister is $530,000 for the canister itself, $110,000 for the storage overpack, and $10,000 for a share of the pad (Stroupe, 1995). Undiscounted fixed costs for the facility, about 57 percent of which would be allocable to the foreign research reactor spent nuclear fuel program, include $24 million for security fencing, other fixed facility costs, licensing, etc.

It is estimated that about 57 percent of the $100 million discounted cost of a wet storage pool constructed from 1996 through 2002 would be allocated to the foreign research reactor spent nuclear fuel program.

Operations and maintenance and safeguards and security costs of about $3.2 million per year are allocated about 57 percent to the foreign research reactor spent nuclear fuel program (WSRC, 1995c). For the wet storage pool, operations and maintenance costs are about $5 million per year higher than at the dry facility. These costs are also allocated about 57 percent to the foreign research reactor spent nuclear fuel program.

The net present value of the allocated expenditures to receive and dry-store the aluminum-based spent nuclear fuel at the Savannah River Site is the sum of the net present values on the bottom row of Table F-127 or about $350 million for dry storage (and about $400 million for wet storage, not shown). Additional expenditures at the Savannah River Site would be incurred to receive and store target material ($20 million) and qualify the aluminum-based spent nuclear fuel for the repository ($25 million). The site-specific components of the other cost factors described in Section F.7.4 (e.g., additional characterization requirements, materials processing prior to repository packaging) would also be extra.

---

7 *Aluminum-based elements can be related according to equivalent MTR elements. The precise loading level is 255 MTR-equivalent elements. Excluding the RHF elements from France, the average aluminum-based foreign research reactor spent nuclear fuel element is equal to 1.12 MTR-equivalents. RHF elements are rated at 20 MTR-equivalents each. The weighted-average MTR-equivalent from the MTR type elements and RHF elements is 21,400. The mass of aluminum-based elements is 101,300 kgTM. This value is based on 85 RHF elements at 110 kg (243 lb) each, 2,650 NRU elements (from Canada) at 5.7 kg (12.6 lb) each, and 15,064 MTR-type elements at 5.1 kg (11.2 lb) each. Target material is excluded from these calculations. Including target material would increase the total to 104,700 kgTM excluding the can.*

F.7.2.7 Storage at the Idaho National Engineering Laboratory

In the base scenarios involving United States acceptance of foreign research reactor spent nuclear fuel, approximately 4,900 TRIGA elements would be shipped to the Idaho National Engineering Laboratory for storage in existing facilities.

The Idaho National Engineering Laboratory would store the TRIGA spent nuclear fuel in the IFSF until the spent nuclear fuel was transferred to canisters for shipping to the repository. Table F-128 shows annual operating costs for the Idaho National Engineering Laboratory to dry-store approximately 4,900 TRIGA elements at the IFSF. (The Idaho National Engineering Laboratory could also wet-store the TRIGA elements at the FAST facility for about twice the cost as at IFSF.) The discounted total cost using the IFSF facility for storage is approximately $30 million. Qualification of TRIGA spent nuclear fuel for repository disposal would add another $15 million.

To complete the transfers from existing storage facilities to repository-qualified dry storage canisters, the Idaho National Engineering Laboratory might eventually require a new staging facility similar to that at the Savannah River Site. The Idaho National Engineering Laboratory is deferring construction of this facility until the repository waste acceptance criteria are available some time after 2000. Based on the share of TRIGA spent nuclear fuel relative to all material to be dry-stored at the Idaho National Engineering Laboratory and geologically disposed, the foreign research reactor spent nuclear fuel program would be allocated no more than $10 million of the capital cost of a staging facility whose discounted total cost would be less than $150 million. Allocable operating costs would be the same as shown in the column for repository canister loading. For cost analysis, repository loading and shipping is assumed to take place in 2030. Actual loading could take place earlier but cannot be specified at present.

F.7.2.8 Chemical Separation at the Savannah River Site

Implementation of Management Alternative 1 (revised to incorporate chemical separation) at the Savannah River Site could take place in different ways. One bounding case is to assume that existing and new facilities are used in essentially the same way as in Management Alternative 1 (storage). RBOF and L-Reactor disassembly basin are used for receiving and lag-storage, a new staging and characterization facility is required for repository loading of aluminum-based material received after the completion of chemical separation operations, one of the Canyons (F- or H-Canyon) is used at a moderate rate, new dry or wet storage facilities are required, etc. This option can be viewed as the separation of foreign research reactor spent nuclear fuel within a larger program to store and directly dispose non-foreign aluminum-based spent nuclear fuel.

The other bounding case can be viewed as the separation of foreign research reactor spent nuclear fuel within an accelerated program to chemically separate all of the accumulated aluminum-based materials and medium-term receipts. In this case, RBOF continues to be used for 40 years for receipts, characterization, storage and repository loading; no new staging and characterization facility is constructed; receipts of aluminum-based spent nuclear fuel from domestic sources are accelerated; one of the Canyons (F- or H-Canyon) is used at an accelerated pace; and no new dry or wet storage facilities are required.

In either case, chemical separation continues to around 2008 to 2010, at which point the canyons are shut down. In the first case, however, enough material remains on site and due to be received that a large-scale storage program (including a new staging and characterization facility) is required. In the second case, very little separable material remains on site and only about 5 casks per year are due to be received at the

Table F-128 Storage of TRIGA Spent Nuclear Fuel at the Idaho National Engineering Laboratory (Millions of 1996 Dollars)

Year	*Capital Costs -Staging*	*Transfers*	*IFSF Operations*	*Repository Canister Loading*	*Repository Canisters*	*Operations & Maintenance*	*Decontamination &Decommissioning*
1996		.4	1				
1997	1	.4	1				
1998		.4	1				
1999		.4	1				
2000		.4	1				
2001		.4	1				
2002		.4	1				
2003		.4	1				
2004		.4	1				
2005		.4	1				
2006		.4	1				
2007		.4	1				
2008		.4	1				
2009			1				
2010			1				
2011			1				
2012			1				
2013	3		1				
2014	4		1				
2015	3		1				
2016-2029			1/yr.				
2030			1	5	10	.1/yr	
2031-2035							2
Total Costs (Undiscounted)	11	5	35	5	10	4	2
NPV	5	4	17	1	2	2	0

time the Canyons are shut down. In this latter case, existing facilities can handle all program functions, including repository canister loading.

The first case is used for cost analysis purposes in this Appendix. This case is more probable, since it is more conservative with respect to selection of separation as an alternative and more conservative with respect to costs.

In either case, uranium (but not plutonium) is chemically separated from fission products at one of the canyons at the Savannah River Site.[8] In this EIS, costs for operations at F-Canyon are used since they are slightly higher than costs at H-Canyon (about $25 million). The credit for recovered uranium is the same in either case.

For either Management Alternative 1 (revised to incorporate chemical separation) (17,800 elements) or Management Alternative 3 (12,200 elements), the following assumptions apply:

---

8 *HEU cannot be chemically separated from LEU. Plutonium can be chemically separated from uranium and fission products but it is not the intention of the Savannah River Site or this EIS to do so. The amount of plutonium in the foreign research reactor spent nuclear fuel is negligible.*

- Basin operations continue until all material can be transferred out of the basins to the Canyons. Foreign research reactor spent nuclear fuel is out of the basins in 2006 under Management Alternative 1 (revised to incorporate chemical separation) and 2005 under Management Alternative 3. From 2003 forward, all receipts take place at the new staging and characterization facility.

- Canyon operations would take place over a maximum of 13 years (1998 through 2010). Actual operations could be completed by about 2009 under Management Alternative 1 (revised to incorporate chemical separation) and as soon as the final shipments were received under Management Alternative 3. The selection of a canyon or canyons will be specified in the Interim Management of Nuclear Materials EIS (DOE, 1995b) and a facilities utilization study currently in process at the Savannah River Site. If no processing is selected in either of those studies, none will be selected for foreign research reactor spent nuclear fuels.

- All Canyon operations that apply to foreign research reactor spent nuclear fuel apply to other, similar materials at the Savannah River Site, i.e., domestic research reactor spent nuclear fuel and DOE and government aluminum-based spent nuclear fuel. Canyon capacity is shared according to the MTR-equivalents of material in each category (foreign and non-foreign) on-site in each year.

- All spent nuclear fuel would be separated incrementally to at-risk materials under the Interim Management of Nuclear Materials EIS (DOE, 1995b).[9] Processing of at-risk materials would be completed in 2002 or early 2003. Spare dissolver capacity would be available for aluminum-based spent nuclear fuels at a rate of 720 MTR-equivalents in 1998 and 1999 and 2880 MTR-equivalents thereafter. Aluminum-based spent nuclear fuel could be separated incrementally over the period 1998 through 2002 or early 2003. Costs (at F- and H-Canyon) for all aluminum-based spent nuclear fuels would be $10.4 million in 1996, $13 million in 1997, $18 million in 1998 and 1999, $5 million in 2000, $3 million in 2001, $13 million in 2002, and $32 million per year thereafter.

- Recovered HEU is blended down to LEU for sale to a commercial power reactor operator. The value of the LEU is assumed to be 85 percent of the value of fresh LEU for power reactors. A value of $5,000 per MTR-equivalent is used for the sales revenue.

- A penalty is assessed on any programs that defer phasedown of Canyon operations beyond the point that the Canyon would be "deinventoried." At some point after 2002, separation of research reactor aluminum-based spent nuclear fuel may become the final base mission at a Canyon. If so, continued operations would incur a deferral penalty estimated at $11 million in the first year of deferral, $22 million in the second year, and $33 million per year thereafter.[10]

Table F-129 shows the costs allocated to the foreign research reactor spent nuclear fuel program for activities at the Savannah River Site under Management Alternative 1 (revised to incorporate chemical

9 *At-risk materials are materials that require stabilization through processing to ensure long-term safety and security.*

10 *A further penalty for deferring "deactivation" is not assessed. The Savannah River Site does not believe that continued operations through about 2008 to 2010 will defer the transition from deinventoried status to deactivated status.*

Table F-129 Chemical Separation Costs at the Savannah River Site Under Management Alternative 1 (Revised to Incorporate Chemical Separation)

Year	*Allocated Receiving, Lag Storage, Facilities Support*	*Spare Dissolver Capacity (MTR-equivalents)*	*Approximate Foreign Research Reactor Share of Dissolver (percent)*	*Allocated Canyon Operations Cost*	*Allocated Canyon Deferral Penalty*	*Allocated HEU Credit*
1996	16	0				
1997	22	0				
1998	25	720	72	13		3
1999	26	720	72	13		3
2000	27	2880	72	4		10
2001	22	2880	72	2		10
2002	23	2880	72	10		10
2003	17	2880	72	23	8	10
2004	16	2880	72	23	16	10
2005	19	2880	72	23	24	10
2006	8	2880	72	23	24	10
2007	0	2880	72	23	24	10
2008	1	2880	72	23	24	10
2009	1	2880	60	19	20	9
2010	1	2880	0	21	20	3
2011-2035	0-2/yr.					
Total Costs (Undiscounted)	242			199	139	107
NPV	178			127	81	70

separation). The foreign research reactor spent nuclear fuel program incurs costs at the percentage shown in the fourth column. This percentage is approximately the spare dissolver capacity allocated to the foreign research reactor program in each year. The change in the percentage at the end occurs because less than proportional dissolver capacity is required to complete the foreign research reactor spent nuclear fuel processing.

Table F-130 shows the same information as Table F-129, adjusted for the shipment of 5,600 aluminum-based elements to Dounreay, Scotland (Management Alternative 3). Table F-130 shows significant cost reductions at the Savannah River Site for basin operations (and related non-processing activities). Separation operations are shown ending in 2007 even though receipts continue through 2008. This is a function of the dissolver capacity available through 2010 and the reduced quantity of foreign research reactor spent nuclear fuel at the Savannah River Site under Management Alternative 3. As a practical matter, the cost impact of stretching out the processing through 2008 is insignificant.

Expenditures at the Savannah River Site for receiving and storing the target material and at the Idaho National Engineering Laboratory for receiving and storing the TRIGA spent nuclear fuel are the same as in the Management Alternative 1 (storage). These values are about $20 million (out of a total of about $35 million) and $50 million, respectively.

F.7.2.9 New Technology

Under Implementation Alternative 7 of Management Alternative 1 (discussed in Section 2.2.2.7 of the EIS), DOE would initiate a development program to select a new treatment and/or packaging technology

Table F-130 Chemical Separation Costs at the Savannah River Site Under Management Alternative 3

Year	*Allocated Receiving, Lag Storage, Facilities Support*	*Spare Dissolver Capacity (MTR-equivalents)*	*Approximate Foreign Research Reactor Share of Dissolver Capacity (Percent)*	*Allocated Canyon Operations Cost*	*Allocated Canyon Deferral Penalty*	*Allocated HEU Credit*
1996	12	0				
1997	17	0				
1998	20	720	62	11		
1999	22	720	62	11		2
2000	22	2880	62	3		2
2001	16	2880	62	2		9
2002	16	2880	62	8		9
2003	13	2880	62	20	7	9
2004	11	2880	62	20	14	9
2005	7	2880	62	20	20	9
2006	0	2880	62	20	20	9
2007	0	2880	6	2	2	1
2008	0	2880	0	0	0	0
2009	0	2880	0	0	0	0
2010	0	2880	0	0	0	0
2011-2035	.1-.5/yr.					
Total Costs (Undiscounted)	173			117	63	68
NPV	129			80	39	47

which would then be constructed and operated to manage the foreign research reactor fuel. A number of different technologies will be considered before one or more are selected for further development.

In addition to the uncertainty as to which technology(ies) will be chosen, there are other cost uncertainties including: the repository disposal fee, the need for new facilities and the requirements needed for managing domestic fuel. To account for these uncertainties, a range of costs have been developed. The costs range from about $950 million (undiscounted) or $625 million (discounted) to about $1.75 billion (undiscounted) or $950 million (discounted).

F.7.2.10 Reprocessing in the United Kingdom

Under Management Alternative 3, approximately 5,600 aluminum-based spent nuclear fuel elements would be shipped to the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland for reprocessing.[11] The remaining 12,200 aluminum-based elements would be chemically separated at the Savannah River Site (Section F.7.2.8).[12] The TRIGA spent nuclear fuel would be stored at the Idaho National Engineering Laboratory (Section F.7.2.7).

---

11 Equal to about 7,900 MTR-equivalents, including 85 RHF elements at 20 MTR-equivalents apiece.

12 Equal to about 13,600 MTR-equivalents, including the 2,650 Canadian NRU elements and all other elements (excluding the French RHF elements) at 1.12 MTR- equivalents apiece.

The number of elements to be reprocessed at Dounreay is based on the number of spent nuclear fuel elements in countries with commercial nuclear power programs and the clear capability to manage the reprocessing wastes.[13] The reprocessing waste from Dounreay is returned to the countries of origin. More generally, Management Alternative 3 can be viewed as chemical separation of approximately 2/3 of the aluminum-based foreign research reactor spent nuclear fuel elements in the United States and 1/3 in the United Kingdom.

Table F-131 shows the United Kingdom Atomic Energy Authority's currently estimated costs to reprocess aluminum-based spent nuclear fuel elements. The cost for conversion assumes a downblending ratio of 2:1 (i.e., one unit of depleted uranium at 0 percent enrichment is added to each unit of separated uranium at 40 percent enrichment to produce two units of uranium at 20 percent enrichment). The costs in Table F-131 are converted from British Pounds to United States Dollars at a rate of 1.55 dollars per pound. Using these costs, the discounted cost to ship, receive, reprocess, and dispose of the wastes from 5,600 aluminum-based spent nuclear fuel elements on a schedule similar to that at the Savannah River Site and to obtain LEU metal fuel is approximately $265 million. At a discount rate of 3 percent, this is equivalent to about $7,000/kgTM, including a charge of about $700/kgTM for blending down the separated HEU to LEU.

Table F-131 Costs at Dounreay (1996 Dollars)

Activity	*Cost*
Transport Casks to/from Dounreay	$31,000/cask @ 2 casks per shipment
Receive & Unload Casks	$7,700/cask
Reprocess and produce cementous intermediate-level waste	$5,750/kgTM (HEU only)
Convert Uranyl Nitrate to Metal	$4,500/kg uranium metal (or $2,800/kg UO_2 for oxide)
Value of Metallic Uranium	$15,000/kg uranium
Store U-235	$1,550/kg U-235
Store intermediate-level waste	$1,550 per 500 l (132 gal) drum per year (containing 10 kg (22 lb) of spent nuclear fuel wastes)
Transport intermediate-level waste to originating country	$2,600/drum
Geologic disposal of intermediate-level waste	$31,000/drum

Source: All Costs (except value of metallic uranium) (Scullion, 1995).

Foreign research reactor operators may prefer to view their costs as the sum of the undiscounted current costs for shipping, reprocessing, and uranyl nitrate conversion to metal (without downblending to LEU) plus the discounted costs for interim storage of uranium, interim storage of reprocessing waste, and geologic disposal of reprocessing waste. Assuming a 3 percent discount rate for the outyear costs, and excluding the value of recovered metal uranium, the reactor operator would estimate a current cost of about $9,500/kgTM, excluding the value of the recovered uranium and $7,200/kgTM including the value of the recovered uranium. At a zero percent discount rate, which is reasonable if the reactor operator wants to incorporate a risk-adjustment for long-term unknowns like geologic disposal, the current costs are about $12,700/kgTM. The value of recovered metal HEU is credited to Dounreay to make it consistent with the value of the recovered LEU at the Savannah River Site. Blend-down at Dounreay would cost about $700/kgTM on a current cost basis. Since these cost estimates are based on current costs (i.e., 1996 dollars in 1996) rather than the current fraction of a series of costs (i.e., 1/13 of 13 years' worth of constant costs over the 1996 through 2008 period at the Savannah River Site), they are exposed to escalation.

---

13 Belgium, France, Germany, Italy, Spain, Switzerland, United Kingdom.

In comparing these costs to costs for managing the spent nuclear fuel in the United States, the key technical differences the reactor operator sees are: 1) receipt of converted fresh metal (with or without downblending), and 2) receipt of cementous waste. How the reactor operator prices receipt of the waste product compared to the charge estimated by the United Kingdom Atomic Energy Authority would have a major impact on the attractiveness of doing business with the United Kingdom versus the United States.

Table F-132 shows the annual cash flows from the European component of Management Alternative 3, including downblending to LEU. Escalation is not included in Table F-132. Net present value is calculated at a 3 percent real discount rate.

Table F-132 Annual Cash Flows from Europe Under Management Alternative 3 (Dollars in Millions)

Year	*Shipping & Receipt*	*Reprocessing*	*Conversion*	*Product Value* [a]	*Store U-235*	*Store Intermediate - Level Waste*	*Transport Intermediate-Level Waste*	*Dispose of Intemediate - Level Waste*
1996	.7	16.6	3.8	6.5	.1	0	0	0
1997	.7	16.6	3.8	6.5	.1	0	0	0
1998	.7	16.6	3.8	6.5	.1	0	0	0
1999	.7	16.6	3.8	6.5	.1	.8	0	0
2000	.7	16.6	3.8	6.5	.1	1.2	0	0
2001	.7	16.6	3.8	6.5	.1	1.5	0	0
2002	.7	16.6	3.8	6.5	.1	1.9	0	0
2003	.7	16.6	3.8	6.5	.1	2.3	0	0
2004	.7	16.6	3.8	6.5	.1	2.7	0	0
2005	.7	16.6	3.8	6.5	.1	3.1	0	0
2006	.7	16.6	3.8	6.5	.1	3.5	0	0
2007	.7	16.6	3.8	6.5	.1	3.8	0	0
2008	.7	16.6	3.8	6.5	.1	4.2	0	0
2009-2025	0	0	0	0	0	4.2/yr	0	0
2026-2030	0	0	0	0	0	0	1.4/yr	16.6/yr
Total Cash Flows (Undiscounted)	10	215	49	84	2	118	8	100
NPV	8	176	40	69	1	65	3	38

[a] *Cost Savings*

F.7.2.11 High-level Waste Vitrification and Separation Waste Storage

The chemical separation operations at the Savannah River Site generate low-level waste and high-level waste that must be managed. Low-level waste is converted to a saltstone material and stored on-site. High-level waste is converted into a borosilicate glass log (vitrification) at the Defense Waste Processing Facility. Costs for the Defense Waste Processing Facility function, including low-level waste handling, are $1.77 million per Defense Waste Processing Facility log. Each log is equal to 300 MTR-equivalents or about 268 typical aluminum-based foreign research reactor spent nuclear fuel elements. The 17,800 aluminum-based elements in the proposed foreign research reactor spent nuclear fuel program equate to about 21,400 MTR-equivalents, including the 85 French RHF elements at 20 MTR-equivalents per element, but excluding target material. This generates 72 Defense Waste Processing Facility logs. Four Defense Waste Processing Facility logs are inserted into one waste package canister (i.e., 18 canisters of glass logs for the alternative). Each canister has an estimated cost of $480,000. Total discounted costs to prepare the high-level waste for geologic disposal are about $65 million. Disposal costs are extra.

F.7.2.12 Transportation to the Repository

The canisters of foreign research reactor spent nuclear fuel or vitrified high-level waste must be shipped to a geologic repository for ultimate disposal. For cost estimation only, the approximate distances from the Idaho National Engineering Laboratory and the Savannah River Site to the candidate repository at Yucca Mountain, Nevada, were used. (No claim regarding the suitability of Yucca Mountain is implied by this assumption.) Using these distances, and assuming shipments by truck, the undiscounted and discounted total costs of repository shipping are estimated at $19 million and $3 million, respectively for all spent nuclear fuel and about $1 million (undiscounted) if the aluminum-based spent nuclear fuel is converted to vitrified high-level waste (EG&G, 1994b).

Shipping costs are sensitive to aluminum-based spent nuclear fuel canister loadings. At approximately 14.4 kg (31.7 lb) U-235 per canister on average, the foreign research reactor spent nuclear fuel program (excluding target material) requires about 602 canisters for aluminum-based spent nuclear fuel and 16 canisters for TRIGA spent nuclear fuel. Reductions in the canister loading translate directly into more shipments. Shipping costs are not highly sensitive to high-level waste disposal packaging. In the base case, converting all the aluminum-based foreign research reactor spent nuclear fuel to vitrified high-level waste generates about 18 canisters.

F.7.2.13 Disposal of Spent Nuclear Fuel

The base case method of disposing of foreign research reactor spent nuclear fuel is intact in poisoned canisters. Preliminary costs to dispose of intact foreign research reactor spent nuclear fuel were developed for DOE's Office of Civilian Radioactive Waste Management in November, 1995 (TRW, 1995). Table F-133 shows the canister loadings, number of canisters, and canister costs (not repository costs) for the internal criticality packaging strategy of 14.4 kg (31.7 lb) U-235 per disposal canister.

Table F-133 Internal Criticality Packaging Strategy: Number Required and Cost Detail (in Thousands of 1996 Dollars)

Fuel Type	*U-235 Max kg/Element*	*Max (kg/package) U-235*	*Max Element/ Package*	*Package Cost ($1000)*	*Actual Element/ Package*	*Number Elements*	*Number Packages*	*Package Cost $1000*
LE MTR	0.42	43	102	114	48	4,838	101	11,490
HE MTR	0.42	14.4	34	92	24	5,692	237	21,819
Dense MTR	0.84	14.4	17	92	24	296	12	1,134
LE Tube	0.40	43	108	114	48	1,149	24	2,729
HE Tube	0.40	14.4	36	114	48	2,928	61	6,954
LE Cluster	0.49	43	88	215[a]	48	1,695	35	7,557
HE Cluster	0.49	14.4	29	114[a]	24	1,097	46	5,211
RHF	9.20	14.4	1	57.1	1	86	86	4,911
LE TRIGA	0.038	43	1132	315[b]	1008	3,834	4	1,198
HE TRIGA	0.133	14.4	108	92[b]	96	1,106	12	1,060
Total						22,721	618	$64,0964

Source: TRW, 1995

[a] *Cost category shifted up 1 to (approximately) account for the fact that the longer length of the Cluster elements precludes their being stacked in three layers (so that the ratio is only 4 to 1).*

[b] *Cost category shifted down by 2 to account for the fact that the smaller width (diameter) of the TRIGA elements permits 16 to fit into the same cross-sectional area as PWR assembly in the commercial waste package.*

The discounted canister-related cost of the packaging strategy displayed in Table F-133 is $11 million. The cost to dispose of the canisters depends on the size of the canisters and the loading levels. An estimate for disposal of full-size (i.e., commercial-type) spent nuclear fuel canisters prepared by the Idaho National Engineering Laboratory equated to $1.8 million per canister in 1994$ in 1994 (Stroupe, 1995), including transportation to the repository. This translated into $2.07 million per canister in 1996$ in 1996, the baseline cost for this EIS. These canisters contained 120 MTR-equivalents of aluminum-based spent nuclear fuel and 500 TRIGA spent nuclear fuel elements.

For the much smaller canisters and lower loading levels shown in Table F-133, a total undiscounted disposal cost (excluding transportation) of $373 million is estimated (TRW, 1995). This translates into an implied charge per canister of approximately $100 thousand for canisters containing aluminum-based spent nuclear fuel and $150 thousand for canisters containing TRIGA spent nuclear fuel. Assuming that repository development costs (1/3 of total repository charges) are incurred from 1996 through 2029 and repository emplacement costs (2/3 of total repository charges) are incurred in 2030 through 2035, the discounted cost of the disposal program (excluding the canisters) is approximately $110 million. About 95 percent of this charge is for disposal of aluminum-based spent nuclear fuel. Discounted total costs for intact disposal of aluminum-based spent nuclear fuel and TRIGA spent nuclear fuel including canister are approximately $125 million.

Total program costs are highly sensitive to the timing of disposal. Accelerating disposal to the 2015 to 2020 time period (rather than 2030 to 2035) reduces undiscounted costs by $50 million but increases discounted costs by $50 million. The savings arise from fewer years of storage prior to repository loading. The discounted cost penalty arises because the large outyear costs for repository development and emplacement lose 15 years of discounting.

F.7.2.14 Disposal of Vitrified High-level Waste

High-level waste is vitrified in the Savannah River Site Defense Waste Processing Facility. The borosilicate glass logs are inserted into waste packages (i.e., metal canisters similar to that used to dispose of commercial spent nuclear fuel) and disposed geologically. The cost to dispose of each waste package is estimated at $1.61 million, including transportation to the repository. At four Defense Waste Processing Facility logs per waste package, the aluminum-based foreign research reactor spent nuclear fuel would generate about 18 waste packages. Discounted disposal costs would be about $10 million. The discounted cost to dispose of the 35 TRIGA spent nuclear fuel canisters at the same loading as described in Section F.7.2.13 is about $10 million. Discounted costs increase by $15 million and undiscounted costs decrease by $25 million due to accelerating repository disposal by 15 years.

F.7.2.15 Storage or Reprocessing Overseas (Management Alternative 2)

For the purpose of the cost analysis, the primary steps in Management Alternative 2 are as follows:

- Each country retains its spent nuclear fuel.
- The countries with commercial nuclear power reprocessing programs reprocess their spent nuclear fuel. The other countries dry-store their spent nuclear fuel.
- Spent nuclear fuel is geologically disposed in an unspecified manner at multiple sites.

Discounted costs for Management Alternative 2 are estimated (very roughly) at $1.25 billion. Costs for this alternative are highly speculative since there is no basis for estimating how most countries would manage their spent nuclear fuel individually or collectively or what types of facilities or approaches they would (or could) select. Of the 41 countries in the proposed foreign research reactor spent nuclear fuel program, 22 have no commercial nuclear power infrastructure to support either a storage or a reprocessing program. These 22, and most of the remaining 19, have no clear program for geologic disposal. Since no country inside or outside the proposed foreign research reactor spent nuclear fuel program has offered to store or dispose of the spent nuclear fuel from other countries, there is no obvious method by which most of the countries in the program could manage their spent nuclear fuel. The costs shown here assume substantial cost penalties from the establishment of up to 22 new spent nuclear fuel storage installations, including all supporting infrastructure.

Reprocessing at the United Kingdom Atomic Energy Authority's facility at Dounreay, Scotland is already an option for Euratom countries that can accept the return of the reprocessing waste. If the United Kingdom Atomic Energy Authority were to reprocess all the material in the foreign research reactor spent nuclear fuel program, including fuels for which it has no current commercial capability, direct costs would exceed $1 billion. Logistics would be highly problematic, however, since the United Kingdom Atomic Energy Authority would require at least 35 years to complete the task at its currently offered capacity. The limited number of other facilities that could reprocess commercial spent nuclear fuel, e.g., the French facility at Marcoule, have not made any commitments to do so. The technical and cost uncertainties associated with disposal of either spent nuclear fuel or high-level waste are entirely speculative but must be considered extremely high.

Overall, there is no basis for assuming that distributed management of the spent nuclear fuel and, in particular, distributed geologic disposal of the spent nuclear fuel or high-level waste, could be accomplished at a cost remotely resembling that of the United States or any other country with a large-scale commercial nuclear power infrastructure.

F.7.3 Interpreting the Minimum Program Costs

Table F-120 (Section F.7.1.2) showed the minimum discounted program costs for the five bounding scenarios. The table showed that for the discount rates appropriate for the U.K. and U.S. portions of the program, hybrid chemical separation/reprocessing of aluminum-based spent nuclear fuel in the United States and the United Kingdom (Management Alternative 3) was about as costly as chemical separation of aluminum-based spent nuclear fuel in the United States alone. Either of the chemical separation/reprocessing approaches was substantially less costly than storing and directly disposing of all the spent nuclear fuel in the United States.

In interpreting the minimum discounted program costs, note that important components of the costs of multiple alternatives are fixed or nearly fixed. Table F-134 shows this relationship. For example, shipping to the United States is the same whether all the spent nuclear fuel is stored or separated. This means that the differences between the costs for the key management function (i.e., storage and disposal or chemical separation and disposal) are substantially larger (in percentage terms) than the differences between the total costs of an implementation alternative. It also means that risks in the unique components of the various implementation alternatives will have an outsized impact on the relative costs of the alternatives.

Table F-134 shows that the undiscounted costs for Management Alternative 1 (storage) exceed $1.4 billion, excluding target material and all other cost and risk factors. The undiscounted costs for Management Alternative 1 (revised to incorporate chemical separation) are approximately $1 billion. Undiscounted costs for Management Alternative 3 are about $1.1 billion. A substantial portion of the cost premium for Management Alternative 3 is due to diseconomies of scale in using the Savannah River Site for two-thirds of the aluminum-based foreign research reactor spent nuclear fuel rather than all of the foreign research reactor spent nuclear fuel. Although not shown in Table F-134, use of a 4.9 percent discount rate for the European component of Management Alternative 3 would generate total costs that are indistinguishable from those under Management Alternative 1 (revised to incorporate chemical separation).

Table F-134 Composition of Minimum Program Costs for Spent Nuclear Fuel Management, 1996 Dollars

	Logistics & Program Management	*Shipping to United States*	*P-1 Operations at SRS*	*P-2 Operation at SRS, including Repository Fuels Qualification*	*P-1/P-2 Operations at INEL, Including Repository Fuels Qualification*	*Ship & Dispose Spent Nuclear Fuel*	*Reprocess Aluminum-Based Spent Nuclear Fuel (Net)*	*Stabilize, Ship & Dispose High-Level Waste*	*Dounreay (Including Blend Down, Net)*	*Total*
Management Alternative 1 (Storage, Dry)[a] Undiscounted	65	194	238	354	92	486	0	0	0	1428
Discounted	47	141	174	195	47	123	0	0	0	727
Management Alternative 1 revised to incorporate Chemical Separation[a] Undiscounted	65	194	190	55	92	13	231	150	0	989
Discounted	47	141	148	31	47	3	138	68	0	623
Management Alternative 1 revised to incorporate a New Technology[b] Undiscounted	65-70	+218	263	222-376	92	123-715	0	0	0	983-1,734
Discounted	47-52	158	191	129-241	47	54-244	0	0	0	626-933
Management Alternative 3[a] Undiscounted	44	124	136	37	92	13	113	100	417	1076
Discounted	32	90	108	21	47	3	73	45	263	682

[a] *No target material*

[b] *Includes target material*

Variations in the share of aluminum-based spent nuclear fuel that is stored/disposed in the United States versus separated/disposed in the United States shift the program costs within the boundary points established in Table F-134 for implementation alternatives to Management Alternative 1. This shift is non-linear, as summarized below:

- Spent Nuclear Fuel Repository Qualification - The proposed foreign research reactor spent nuclear fuel program would be responsible for the costs to repository-qualify an estimated three types of aluminum-based spent nuclear fuel. The discounted cost to characterize these three fuel types is approximately $25 million. (See Table F-123). If all the aluminum-based foreign research reactor spent nuclear fuel were separated, there would be no repository qualification and no charge to the program. Whether separation of less than all the aluminum-based spent nuclear fuel eliminates one or more fuel types from qualification requirements would depend on when the fuel was received, where each fuel type appeared on the prioritization for separation, and how long separation continues.

- Canyon Operating Costs - Canyon operating costs allocated to the foreign research reactor spent nuclear fuel program are at a minimum during the years when processing is incremental to processing under the Interim Management of Nuclear Materials EIS (1998 to 2002) and higher afterwards. Switching from incremental costing to average variable costing increases annual costs from as little as $1 million for 2,880 MTR-equivalents to about $32 million. Including the phase-down penalty (Section F.7.2.8) increases the cost by approximately $33 million per year. The timing of the switch from incremental costing to average variable costing (and thus the impact on the foreign research reactor spent nuclear fuel program) depends on decisions made under the Interim Management of Nuclear Materials EIS (DOE, 1995b) and a facilities utilization study underway at the Savannah River Site. The timing of any deferral penalty is subjective. It depends on whether other missions for the Canyons have been identified and whether plans to deinventory the Canyon used by the foreign research reactor spent nuclear fuel program have been developed. It is clear that Canyon operations costs allocable to the foreign research reactor spent nuclear fuel program per year or per MTR-equivalent would be much higher after 2002 than before 2002 but it is not certain how much higher or when they would become higher. This uncertainty prevents a linear estimation of separation costs according to the quantity of material processed. Section F.7.4 discusses this issue in more detail.

- Staging and Characterization Facility Capital Costs -- The Savannah River Site plans to construct a staging facility to transfer spent nuclear fuel from the existing wet basins to interim dry storage and ultimately to repository canisters. The unallocated discounted capital cost of this facility exceeds $150 million. There is no necessarily correct way to allocate the capital costs of this facility since it supports multiple components of multiple programs and is sized according to joint requirements of multiple programs. Section F.7.2.6 described the cost allocation approach used in this EIS. Approaches that could increase the costs allocated to the foreign research reactor spent nuclear fuel program are also plausible.

- Basin Operating Costs -- The Savannah River Site has estimated the costs to operate RBOF and L-Reactor disassembly basin over a roughly 10-year period at a round-the-clock operations level but has no generalized relationship that permits continuous variation in basin costs according to the number of elements received or stored. Costs depend on the timing of the receipts, the amount of characterization and canning, intra-site and inter-site transfer requirements, the variability in year-to-year staffing, and other factors.

Section F.7.4 outlines four additional groups of factors of significance in using the minimum program costs in Table F-120 as a decision basis for the program.

F.7.4 Interpreting the Other Cost Factors

Table F-120 showed the minimum discounted cost for the five bounding scenarios. The costs in Table F-120 include component contingencies but they do not include system risks, component and non-component risks, or the effects of discount rate changes.[14] Table F-121 showed these latter factors for the five scenarios. Detailed discussions are presented below. Real escalation is excluded from all costs in both tables.

F.7.4.1 Systems Integration and Logistics

The minimum program costs include the contingencies related to individual components of the program, e.g., shipping, basin operations, storage, transfers, and disposal. The minimum program costs do not include systems integration or logistics risks. The proposed foreign research reactor spent nuclear fuel program involves 41 foreign countries (a majority of which have no commercial nuclear power program), dozens of foreign ports, 13 years of receipts, up to 10 domestic ports, as many as 250 cross-country spent nuclear fuel shipments, at least two management sites, and developmental technologies (especially repository disposal technologies). Substantial systems integration bottlenecks could arise in many technical areas, e.g., insufficient casks to ship at the required rate or at the estimated loadings; vulnerability-related shutdowns at existing facilities; requirements for on-site canning prior to cask loading; unplanned requirements for dry storage characterization or conditioning; unexpected facilities requirements for meeting the repository waste acceptance criteria; delays in repository acceptance; and so forth, including normal project (not component) contingencies.

Substantial bottlenecks could also arise in many procedural areas, e.g., incompatibilities with Naval programs at the Idaho National Engineering Laboratory; requirements for on-site inspections by the International Atomic Energy Agency; constraints on shipments, duration of shipments, shipment routes, or quantities of materials shipped pursuant to agreements with the states, and so forth. Because the list of technical and procedural issues that could delay and complicate the program is both long and highly plausible, it is realistic to expect costs to increase above the component-level minimums that make up Table F-120. This risk is estimated at 10 to 15 percent of minimum discounted program costs.

F.7.4.2 Program Component Risks

Several key components of the foreign research reactor spent nuclear fuel program are uncertain. This section discusses the most important probability-adjusted uncertainties (risks).

- ***The method of disposal of spent nuclear fuel-*** The base case assumption is that aluminum-based HEU spent nuclear fuel and HEU TRIGA spent nuclear fuel can be loaded into poisoned canisters and disposed at the equivalent of 14.4 kg (31.7 lb) U-235 per canister. This packing density could be unacceptable to the repository program. Processing the uranium into an isotopically neutral mass (1 percent U-235) would require construction of a new melt-and-dilute facility. Construction and operation of this facility

---

14 Contingencies refer to costs that are certain to occur based on historical experience with programs of similar maturity. These costs are grouped under the term "contingency" because they cannot be line-itemized. Uncertainties refer to changes in the costs of individual components or the overall program that might occur due to unknown changes in regulations, technical conditions, operational status, etc. They are assigned a probability based on their likelihood. Thus, contingencies will occur--they just cannot be line-itemized; uncertainties may occur-- they are adjusted for their probability of occurrence and expressed as risks.

could add $100 million or more to the cost of spent nuclear fuel disposal. Processing the spent nuclear fuel to avoid severe mass limitations on disposal is considered a high probability event.

- ***The adequacy of limited characterization of the spent nuclear fuel*** - There is technical uncertainty about the requirements for characterizing and conditioning the spent nuclear fuel before storing it. At the Savannah River Site, the characterization stage consists of checking the history of the spent nuclear fuel and its paperwork (documentation), visual inspection, gamma scanning (to verify the presence and amount of fissile material), and a leak detection test ("sipping") to determine if any fission products are escaping from the spent nuclear fuel elements. Canning would be limited to degraded elements only. If more extensive characterization and canning is required, new hot cells may be required. Allocable discounted costs to add and operate a hot cell at the staging facility are on the order of $100 million. The requirement for additional characterization and conditioning is a moderately probable event.

- ***Bottlenecks at the Defense Waste Processing Facility*** - Complete separation of aluminum-based spent nuclear fuel at the Savannah River Site generates about 72 Defense Waste Processing Facility logs at a cost of $1.77 million per log. The Savannah River Site estimates that for capital costs of about $100 million and operating costs of about $40 million per year, it could remove bottlenecks at the Defense Waste Processing Facility such that the cost would decline to $1.0 million per year. For the foreign research reactor spent nuclear fuel program, the allocated cost of the capital and operating requirements to relieve the bottleneck is a few million dollars. The discounted savings would be in the range of $50 million. The likelihood that the foreign research reactor spent nuclear fuel program would realize these savings is low to moderate.

- ***Failure to commercially sell the recovered uranium*** - The Savannah River Site might not be allowed to blend-down the recovered HEU for sale as power reactor fuel. DOE, for example, could choose to safeguard the HEU and isolate its chemical separation operations from the commercial power market. This would cost the foreign research reactor spent nuclear fuel program an additional $70 million. The likelihood that the foreign research reactor spent nuclear fuel program would fail to recover this value is low.

F.7.4.3 Non-Program Risks

The key non-program risk is that the cost of repository disposal increases across the board due to a change in scope (not due to escalation within the existing scope). The repository cost allocation used in this EIS assumes no monitored retrievable storage and one geologic repository. If either of these assumptions is incorrect, the cost of the repository component of the program would increase by about 20 percent. If both are incorrect, the cost of the repository component of the program would increase by about 40 percent. These increases translate into cost increases for geologic disposal of intact foreign research reactor spent nuclear fuel of about 5 to 10 percent. The cost of the chemical separation alternative (including disposal of TRIGA spent nuclear fuel) would increase by about 1 to 2 percent.

Cost escalation in the base repository program would also increase the allocated costs for the foreign research reactor spent nuclear fuel program. This type of cost escalation is highly speculative but could be in the tens of millions of dollars for the storage alternatives. Escalation is treated separately from other cost risks.

A second non-program risk is that one or more of the EISs that relate to materials management and facilities use at the Savannah River Site or the Idaho National Engineering Laboratory (besides the foreign research reactor spent nuclear fuel EIS) leads to legal or regulatory action that delays all site activities and throws the foreign research reactor spent nuclear fuel program off-schedule or out of the planned facilities.

F.7.4.4 Discount Rates

This EIS uses the real discount rate specified by the Office of Management and Budget for long-term government projects evaluated in the year ending February 1996 (OMB, 1995). The specified rate, 4.9 percent, is historically high. It compares to Office of Management and Budget rates of 3.8 percent, 4.5 percent, and 2.9 percent for the years ending in February of 1993, 1994, and 1995, respectively (OMB 1992; OMB, 1993; OMB, 1994). It also compares to measured real, long-term government interest rates of 3.2 percent, 2.9 percent, 4.1 percent, and 3.4 percent (through 1995 quarter 2), respectively for the years 1992, 1993, 1994, and 1995 (FRB Cleveland, 1995). Finally, it compares to a Congressional Budget Office estimate of 2 percent for government projects independent of the period and duration (Hartman, 1990).

Unlike the United States, the United Kingdom issues some debt instruments that are the equivalent of inflation-adjusted treasury securities. In recent years, these have yielded between 2 and 5 percent. The rate as of mid January, 1996 was approximately 3.6 percent. The United Kingdom is also currently considering the required discount rate (i.e., real rate of return) on trust funds to provide for decommissioning commercial nuclear power plants. Although no decision has been reached, the government supports a 6 percent rate while the United Kingdom nuclear utilities support a 2 percent rate.

The appropriate discount rate for the analysis is the risk-free rate at which funds received today can be invested to cover future expenses. Since receipt of the revenues precede expenditures, a conservative rate is low. This is the reverse of the more common situation where a high rate is used to discount future receipts compared to current expenses. Moreover, where fixed and certain revenues precede variable and uncertain expenses, the need for a conservative (i.e., low) discount rate is even greater. At a 3 percent discount rate, the discounted cost of Management Alternative 1 (storage) increases by $175 million (to $900 million). The cost of Management Alternative 1 (revised to incorporate chemical separation) increases by $120 million. The effect on the storage alternative is much greater because the high out-year costs for repository canisters and repository emplacement are much more prominent at the lower discount rate.

References

AECLT (Atomic Energy of Canada Ltd. Technologies, Inc.), 1994, Dry Spent Fuel Storage Capabilities and Experience.

ANSI (American National Standards Institute), 1984a, *Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)*, ANSI/ANS 57.9-1984.

ANSI (American National Standards Institute), 1984b, *Criticality Safety Criteria for the Handling, Storage, and Transportation of LWR Fuel Outside Reactors*, ANSI/ANS 8.17-1984.

ANSI (American National Standards Institute), 1983, *Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors*, ANSI/ANS 8.1-1983, Revision of ANSI N16.1-1975, October 7.

ANSI (American National Standards Institute), 1975a, *Guide for Nuclear Criticality Safety in the Storage of Fissile Materials*, ANSI N16.5-1975, April 12.

ANSI (American National Standards Institute), 1975b, *Design Objectives for Highly Radioactive Solid Material Handling and Storage Facilities in a Reprocessing Plant*, ANSI N305-1975, March 19.

Australia, Commonwealth of, 1993, *Future Reaction: Report of the Research Reactor Review*, August.

Bergsman, K. H., A. N. Praga and W. L. Willis, 1994, (Westinghouse Hanford Company), *Hanford Site Technical Information for Foreign Research Reactor Fuel Environmental Impact Statement*, August 26.

BG&E (Baltimore Gas and Electric Company), 1989, *Calvert Cliffs Independent Spent Fuel Storage Installation Safety Analysis Report*, Baltimore, MD, December.

BNFL (British Nuclear Fuels, Ltd.), 1994a, *Waste Management and Storage*, Risley, Warrington, Cheshire, United Kingdom.

BNFL (British Nuclear Fuels, Ltd.), 1994b, *Reprocessing Seminar for United States Utilities*, Risley, Warrington, Cheshire, United Kingdom, March.

Bonser, D. R., 1994, "Reprocessing in THORP - Today and for the Future," *RECOD '94*, April 24-28.

Cartwright, P., 1994, United Kingdom Atomic Energy Authority, personal communication to A. Murray, Science Applications International Corporation, November 8.

Chandler, G., D. Collins, K. Dye, C. Eberhart, M. V. Hynes, R. Kovach, R. Ortiz, J. Perea, and D. Sherman, 1992, *Assessment of the Facilities on Jackass Flats and other Nevada Test Site Facilities for the New Nuclear Rocket Program*, LA-UR-92-2417, Los Alamos National Laboratory, NM, July.

Claxton, R. E., R. L. Geddes, J. D. Lengel, J. D. Logue, D. C. Losey, R. S. Matthews, D. L. McWhorter, A. P. Poon, W. G. Smith, K. M. Vashi, D. W. Nelson, J. M. Owen, and C. G. Phillips, 1993, *Technical Data Summary Supporting the Spent Nuclear Fuel Environmental Impact Statement*, Rev. 1, NMP-PLS-930182, Westinghouse Savannah River Company, Aiken, SC, November.

COGEMA, 1994a, *Nuclear Materials Reprocessing and Recycling*, La Hague, France, November.

COGEMA, 1994b, *The Reprocessing Division*, La Hague, France, April.

Cosimi, R., 1994, NTS-Reynolds, personal communication to A. P. Murray, Science Applications International Corporation, September 29.

Croff, A. G., 1980, *ORIGEN2 - A Revised and Updated Version of the Oak Ridge Isotope Generation and Depletion Code*, ORNL-5621, Oak Ridge National Laboratory, Oak Ridge, TN, July.

Dahlke, H. J., D. A. Johnson, J. K. Rawlins, D. K. Searle, and G. W. Wachs, 1994, *Summary Engineering Description of Underwater Fuel Storage Facility for Foreign Research Reactor Spent Nuclear Fuel*, EGG-ME-11466, EG&G Idaho, Inc., Idaho Falls, ID, October.

Denney, R., 1994a, Lockheed Idaho Technologies Company, personal communication with H. Chernoff, Science Applications International Corporation, December.

Denney, R., 1994b, Westinghouse Idaho Nuclear Company, personal communication to A. Zeitoun, Science Applications International Corporation, July 15.

DiNunno, J. J., F. D. Anderson, R. E. Baker, and R. L. Waterfield, 1962, *Calculation of Distance Factors For Power And Test Reactor Sites*, TID-14844, March.

DOE (U.S. Department of Energy), 1995a, *Final Programmatic Environmental Impact Statement for Tritium Supply and Recycling*, DOE/EIS-0161, Office of Reconfiguration, October.

DOE (U.S. Department of Energy), 1995b, *Final Environmental Impact Statement, Interim Management of Nuclear Materials*, DOE/EIS-0220, October.

DOE (U.S. Department of Energy), 1995c, *Final Environmental Impact Statement, Safe Interim Storage of Hanford Tank Wastes, Hanford Site, Richland, Washington*, DOE/EIS-0212, Richland Operations Office, October.

DOE (U.S. Department of Energy), 1995d, *Draft Environmental Impact Statement, Management of Spent Nuclear Fuel from the K Basins at the Hanford Site, Richland, Washington*, DOE/EIS-0245D, Richland, WA, October.

DOE (U.S. Department of Energy), 1995e, *Disposition of Surplus Highly Enriched Uranium Draft Environmental Impact Statement*, DOE/EIS-0240-D, Office of Fissile Materials Disposition, October.

DOE (U.S. Department of Energy), 1995f, *Savannah River Site Waste Management Final Environmental Impact Statement*, DOE/EIS-0217, Savannah River Operations Office, Aiken, SC, July.

DOE (U.S. Department of Energy), 1995g, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final Environmental Impact Statement*, DOE/EIS-0203-F, Office of Environmental Management, Idaho Operations Office, April.

DOE (U.S. Department of Energy), 1994a, *Final F-Canyon Plutonium Solutions Environmental Impact Statement*, DOE/EIS-0219, Savannah River Site, Aiken, SC, December.

DOE (U.S. Department of Energy), 1994b, *Multi-Purpose Canister System Preliminary Environmental Evaluation*, DOE/RW-0452, Office of Civilian Radioactive Waste Management, Washington, DC, November.

DOE (U.S. Department of Energy), 1994c, *Notice of Intent to Prepare an Environmental Impact Statement for the Fabrication and Deployment of a Multi-Purpose Canister-Based System for the Management of Civilian Spent Nuclear Fuel, Federal Register* Volume 59, No. 204, (59 FR 53442), October 24.

DOE (U.S. Department of Energy), 1994d, *DOE HANDBOOK: Airborne Release Fractions/Rates and Respirable Fractions for Nonreactor Nuclear Facilities*, DOE-HDBK-3010-94, Washington, DC, October.

DOE (U.S. Department of Energy), 1994e, *Control and Accountability of Nuclear Materials*, Order 5633.3B, September 7.

DOE (U.S. Department of Energy), 1994f, *Multi-Purpose Canister System Evaluation - A Systems Engineering Approach*, DOE/RW-0445, Office of Civilian Radioactive Waste Management, Washington, DC, September.

DOE (U.S. Department of Energy), 1994g, *Draft Supplemental Environmental Impact Statement - Defense Waste Processing Facility*, DOE/EIS-0082-S-D, Savannah River Site, Aiken, SC, August.

DOE (U.S. Department of Energy) 1994h, *Department of Energy Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Draft Environmental Impact Statement*, DOE/EIS-0203-D, Office of Environmental Management, Idaho Operations Office, June.

DOE (U.S. Department of Energy), 1994i, *Environmental Assessment of Urgent-Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel*, DOE/EA-0912, Washington, DC, April.

DOE (U.S. Department of Energy), 1993a, 10 CFR Part 835, "Occupational Radiation Protection; Final Rule," *Federal Register*, Volume 58, No. 238, December 14.

DOE (U.S. Department of Energy), 1993b, *Spent Fuel Working Group Report on Inventory and Storage of the Department's Spent Nuclear Fuel and other Reactor Irradiated Nuclear Materials and their Environmental, Safety and Health Vulnerabilities*, Volumes II and III, November.

DOE (U.S. Department of Energy), 1993c, *Defense Programs Safety Survey Report*, Volume 1: Main Report, DOE/DP170056-H, Office of Defense Programs, November.

DOE (U.S. Department of Energy), 1992a, *DOE STANDARD: Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports*, DOE-STD-1027-92, December.

DOE (U.S. Department of Energy), 1992b, *Technical Safety Requirements*, Order 5480.22, February 15.

DOE (U.S. Department of Energy), 1990, *Radiation Protection of the Public and the Environment*, Order 5400.5, February 8.

DOE (U.S. Department of Energy), 1989a, *General Design Criteria*, Order 6430.1A, April 6.

DOE (U.S. Department of Energy), 1989b, *Environment, Safety, and Health Program for Department of Energy Operations*, Order 5480.1B, March 27.

DOE (U.S. Department of Energy), 1988, *General Environmental Protection Program*, Order 5400.1, November 9.

Duke Power Company, 1988, *Independent Spent Fuel Storage Installation Safety Analysis Report, Oconee Nuclear Station.*

DuPont (E. I. duPont de Nemours & Co.), 1983a, *Safety Analysis - 200 Area Savannah River Plant, Separations Area Operations, Receiving Basin for Offsite Fuel (Sup. 3)*, DPSTSA-200-10-3, September.

DuPont (E. I. duPont de Nemours & Co.), 1983b, *Systems Analysis - 200 Area Savannah River Plant Receiving Basin for Offsite Fuel*, DPSTSY-200-4, Rev. 1, March.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1994a, *Waste Management Facilities Cost Information for Transportation of Radioactive and Hazardous Materials*, EGG-WM-10877, Revision 1, September.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1994b, *Feasibility Study for a DOE Research and Production Fuel Multipurpose Canister*, Idaho National Engineering Laboratory, EGG-WM-11011, February.

EG&G (Edgerton, Germeshausen, and Grier) Idaho, Inc., 1993, *Waste Management Facilities Cost Information Estimating Data for Spent Nuclear Fuel*, EGG-WM-10670 and 10708, March.

Einziger, R. E. and J. A. Cook, 1985, "Behavior of Breached Light Water Reactor Spent Fuel Rods in Air and Inert Atmosphere at 229^{o}C," *Nuclear Technology*, Volume 69, No. 1, pp. 55-71, April.

Elder, J. C., J. M. Graf, J. M. Dewart, T. E. Buhl, W. J. Wenzel, L. J. Walker, and A. K. Stoker, 1986, *A Guide to Radiological Accident Considerations for Siting and Design of DOE Nonreactor Nuclear Facilities*, LA-10294-MS, Los Alamos National Laboratory, NM, January.

EPA (U.S. Environmental Protection Agency), 1991, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents,* Office of Radiation Programs, EPA 400-R-92-001, Washington, DC, October (Reprinted May 1992).

EPRI (Electric Power Research Institute), 1993, *Comparative System Economics of Concrete Casks for Spent-Fuel Storage,* TR-102415, Palo Alto, CA, June.

Fields, G., 1994, Allied General Nuclear Services, personal communication to A. P. Murray, Science Applications International Corporation, June 29.

Fort St. Vrain, 1992, Independent Spent Fuel Storage Installation (ISFSI) Safety Analysis Report.

FRB Cleveland (Federal Reserve Bank of Cleveland), 1995, *Economic Trends,* November.

Gertz, C. P., 1994, U.S. Department of Energy, Nevada Operations, memo to C. R. Head, U.S. Department of Energy, September 28.

Hartman, R. W., 1990, "One Thousand Points of Light Seeking A Number: A Case Study of CBO's Search for a Discount Rate Policy," in *Journal of Environmental Economics and Management,* Volume 18, pp. S3-S7.

Hunter, C. H. and J. Stewart, 1994, *Air Dispersion Modeling for the Spent Nuclear Fuel Environmental Impact Statement - Nonradiological Emissions (U),* WSRC-RP-147 (Draft), Westinghouse Savannah River Company.

Hynes, M. V., 1994, Los Alamos National Laboratory, personal communication to A. P. Murray, Science Applications International Corporation, August 30.

ICRP (International Commission on Radiological Protection), 1991, *1990 Recommendations of the International Commission on Radiological Protection,* Annals of the ICRP, Publication 60, Volume 21, No. 1-3, Pergamon Press, New York, November.

ICRP (International Commission on Radiological Protection), 1979-1982, *Limits for Intakes of Radionuclides by Workers,* Annals of the ICRP, Publication 30, Pergamon Press, New York, November.

ICRP (International Commission on Radiological Protection), 1977, *Recommendations of the International Commission on Radiological Protection,* Annals of the ICRP, Publication 26, Pergamon Press, New York, November.

Jackson, W. N., 1994, Westinghouse Savannah River Company, personal notes from a meeting at the Savannah River Site on the Foreign Research Reactor Spent Nuclear Fuel EIS, May 13.

Johnson, A. B., Jr. and E. R. Gilbert, 1983, *Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases,* PNL-4835, Pacific Northwest Laboratory, September.

Kohli, R., D. Stahl, V. Pasupathi, A. B. Johnson, Jr., and E. R. Gilbert, 1985, "The Behavior of Breached Boiling Water Reactor Fuel Rods on Long-Term Exposure to Air and Argon at 598°K," *Nuclear Technology,* Volume 69, No. 2, pp. 186-197, May.

LaGrange, R., 1995, U.S. Department of Energy, Savannah River Site, personal communication to C. Head, U.S. Department of Energy, February 7.

Levy, I. S., B. A. Chin, E. P. Simonen, C. E. Beyer, E. R. Gilbert, and A. B. Johnson, Jr., 1987, *Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy-Clad Fuel Rods in Inert Gas,* PNL-6189, Pacific Northwest Laboratory, May.

Matthews, R. S., 1994, Westinghouse Savannah River Company, personal communication to A. P. Murray, Science Applications International Corporation, June 29.

Matthews, R. S., 1991, personal communication to E. P. Maddux, NMP-SPA-910150, August 6.

Miller, J. A., J. H. Knick, and S. W. O'Rear, 1995, *Savannah River Site Spent Nuclear Fuel Near Term Strategic Plan*, EFR-950061, May 22.

MMES (Martin Marietta Energy Systems), 1994, *White Paper Summary Report: Spent Nuclear Fuels Management Complex Siting Study*, Oak Ridge, TN, March 18.

Morrison, J., 1994, Westinghouse Savannah River Company, personal communication to H. Chernoff, Science Applications International Corporation, November 30.

Napier, B. A., R. A. Peloquin, D. L. Strenge, and J. V. Ramsdell, 1988, *GENII - The Hanford Environmental Radiation Dosimetry Software System*, PNL-6584, UC-600, Pacific Northwest Laboratory, December.

Neuhauser, K. S. and F. L. Kanipe, 1993, *RADTRAN 4 Volume II: Technical Manual,* SAND89-2370, Sandia National Laboratory, August.

NRC (U.S. Nuclear Regulatory Commission), 1993, "Licensing Requirements For Independent Storage Of Spent Nuclear Fuel And High-Level Radioactive Waste," Title 10 *Code of Federal Regulations*, Part 72.

NRC (U.S. Nuclear Regulatory Commission), 1989, *Safety Evaluation Report Related to the Topical Report for the TN-24 Storage Cask Submitted by Transnuclear, Inc.*, Office of Nuclear Material Safety and Safeguards, July.

NRC (U.S. Nuclear Regulatory Commission), 1988a, *Safety Evaluation Report Related to the Topical Safety Analysis Report for the NAC-C28 Storage/Transport Cask Submitted by Nuclear Assurance Corporation*, Office of Nuclear Material Safety and Safeguards, September.

NRC (U.S. Nuclear Regulatory Commission), 1988b, *Nuclear Fuel Cycle Facility Accident Analysis Handbook,* NUREG-1320, Division of Industrial and Medical Nuclear Safety, Office of Nuclear Material Safety and Safeguards, Washington, DC, May.

NRC (U.S. Nuclear Regulatory Commission), 1987, *Safety Evaluation Report Related to the Topical Safety Analysis Report for MC-10 Dry Spent Fuel Storage Cask Submitted by Westinghouse Electric Corporation*, Office of Nuclear Material Safety and Safeguards, September.

NRC (U.S. Nuclear Regulatory Commission), 1984, *Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation*, Regulatory Guide 3.54, Office of Nuclear Regulatory Research, September.

NRC (U.S. Nuclear Regulatory Commission), 1981, *Design of an Independent Spent Fuel Storage Installation (Water-Basin Type)*, Regulatory Guide 3.49, Office of Nuclear Regulatory Research, December.

NRC (U.S. Nuclear Regulatory Commission), 1979a, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Uranium Fuel Fabrication Plant*, Regulatory Guide 3.34, Revision 1, Office of Standards Development, July.

NRC (U.S. Nuclear Regulatory Commission), 1979b, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Plutonium Processing and Fuel Fabrication Plant*, Regulatory Guide 3.35, Revision 1, Office of Standards Development, July.

NRC (U.S. Nuclear Regulatory Commission), 1977a, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,* Regulatory Guide 1.109, Revision 1, Office of Standards Development, October.

NRC (U.S. Nuclear Regulatory Commission), 1977b, *Assumptions Used for Evaluating the Potential Radiological Consequences of Accidental Nuclear Criticality in a Fuel Reprocessing Plant*, Regulatory Guide 3.33, Office of Standards Development, April.

NRC (U.S. Nuclear Regulatory Commission), 1975, *Spent Fuel Storage Facility Design Basis*, Regulatory Guide 1.13, Revision 1, Office of Standards Development, December.

Nuclear Engineering International, 1993, *World Nuclear Industry Handbook*, Surrey, England.

Nuclear Fuel, 1994a, "SMUD Expects to Recoup Dry Storage Investment In Less Than Two Years," Volume 19, No. 24, pp. 10-11, November 21.

Nuclear Fuel, 1994b, "Cogema Wants To Illegally Store German Spent Fuel, Greenpeace Claims," Volume 19, No. 16, pp. 16-17, August 1.

Nuclear Fuel, 1993, "EDF Postpones Spent Fuel Storage Project, Will Use La Hague Pools," Volume 18, No. 21, p. 9, October 11.

NUTECH (NUTECH, Inc.), 1988, *Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS-24P*, NUH-002, Revision 1, July.

OMB (Office of Management and Budget), 1994, Circular No. A-94, Revised, Washington, DC, February 10.

OMB (Office of Management and Budget), 1993, Circular No. A-94, Revised, Washington, DC, January.

OMB (Office of Management and Budget), 1992, Circular No. A-94, Washington, DC, January.

O'Rear, S. W., 1995, Westinghouse Savannah River Company, personal communication with M. E. Dupont, Westinghouse Savannah River Company, March 8.

O'Rear, S. W., 1994, Westinghouse Savannah River Company, personal communication with H. Chernoff, Science Applications International Corporation, November 29.

Reed, W. C., 1994, EG&G Idaho, Inc., personal communication to A. P. Murray, Science Applications International Corporation, September 30.

Ridal, A. and P. Bull, 1994, *Spent HIFAR Fuel Elements Behaviour Under Extended Dry Storage*, Australian Nuclear Science and Technology Organisation, September.

RSIC (Radiation Shielding Information Center), 1990, *ZYLIND - An Interactive Point Kernel Program For Photon Dose Rate Prediction Of Cylindrical Source/Shield Arrangements*, CCC-557, Oak Ridge National Laboratory.

Scullion, A., 1995, United Kingdom Atomic Energy Authority, personal communication to H. Chernoff, Science Applications International Corporation, October 18.

Shedrow, C. B., 1994a, *Characterization of Proposed SRS Dry Storage Facilities to be Evaluated for the Foreign Research Reactor Spent Nuclear Fuel EIS (U),* ESH-NEP-94-0078, Revision 1, Westinghouse Savannah River Company, June 27.

Shedrow, C. B., 1994b, *Response to Data Call for the Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement,* Draft, Revision 1, Westinghouse Savannah River Company, January 31.

Shirai, E., M. Adachi, T. Kodaira, and H. Sato, 1991, "Dry Storage Facility of JRR-3 Spent Fuels," Third International Conference on Nuclear Fuel Reprocessing and Waste Management, *RECOD '91*, Sendai, Japan, April 14-18.

Silver, J. M., 1993, Australian Nuclear Science and Technology Organisation, personal communication to C. R. Head, U.S. Department of Energy, June 23.

Stroupe, E. P., 1995, Lockeed Idaho Technologies Company, personal communication to J. H. Boyd, U.S. Department of Energy, Idaho Operations Office, October 11.

Taylor, B. K., P. B. Parks, W. E. Bickford, and R. W. Begamins, 1994, *The DOE Information Call on Facilities for Storage, Characterization, and Treatment of DOE-Owned Spent Nuclear Fuel (U)*, WSRC-RP-94-233, Westinghouse Savannah River Company, May.

TRW (TRW Environmental Safety Systems Inc.), 1995, *Assessment of Technical Issues and Cost for Direct Disposal of Foreign Research Reactor Fuel,* A00000000-01717-5705-00010 Rev. 0, December.

Williams, A., 1994, Science Applications International Corporation, personal communication to A. P. Murray, Science Applications International Corporation, June 29.

WINCO (Westinghouse Idaho Nuclear Company, Inc.), 1994, *Assessment of Aircraft Impact Probabilities at the Idaho Chemical Processing Plant,* WINCO-1137, Rev. 1, UC-620, June.

WSRC (Westinghouse Savannah River Company), 1995a, *Savannah River Site Evaluation of Spent Nuclear Fuel Options (U),* WSRC-RP-95-798, November.

WSRC (Westinghouse Savannah River Company), 1995b, *Basis for Interim Operation (BIO) for the L-Reactor in Cold Shutdown (U),* WSRC-TR-95-0054, Aiken, SC, July.

WSRC (Westinghouse Savannah River Company), 1995c, *Basis for Interim Operation (BIO) for the Receiving Basin for Offsite Fuels (RBOF) and the Resin Regeneration Facility (RRF) (U),* WSRC-TR-94-0287, Revision 1, Aiken, SC, March.

WSRC (Westinghouse Savannah River Company), 1994, *Foreign Research Reactor Spent Nuclear Fuel Interim Dry Storage (U),* 000:EPD-SE-94-2208, Aiken, SC, March 31.

WSRC (Westinghouse Savannah River Company), 1992a, *Barnwell Nuclear Fuel Plant Acquisition Considerations - Part I: Executive Summary, History, and Background,* NMP-PLS-920069, Aiken, SC.

WSRC (Westinghouse Savannah River Company), 1992b, *Barnwell Nuclear Fuel Plant Acquisition Considerations - Part II: General Description, Potential Uses, Issues Regarding Acquisition, and Proposed Path Forward,* NMP-PLS-920078, Aiken, SC.

WSRC (Westinghouse Savannah River Company), 1992c, *Barnwell Nuclear Fuel Plant Acquisition Considerations - Part III: Backup Information,* NMP-PLS-920079, Aiken, SC.

WSRC (Westinghouse Savannah River Company), 1992d, *Barnwell Nuclear Fuel Plant Acquisition Evaluation - Phase 1 Exhibits,* NMP-PLS-920083, Aiken, SC.

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix G
Background Documents

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

THE WHITE HOUSE

Office of the Press Secretary

For Immediate Release September 27, 1993

FACT SHEET
NONPROLIFERATION AND EXPORT CONTROL POLICY

The President today established a framework for U.S. efforts to prevent the proliferation of weapons of mass destruction and the missiles that deliver them. He outlined three major principles to guide our nonproliferation and export control policy:

-- Our national security requires us to accord higher priority to nonproliferation, and to make it an integral element of our relations with other countries.

-- To strengthen U.S. economic growth, democratization abroad and international stability, we actively seek expanded trade and technology exchange with nations, including former adversaries, that abide by global nonproliferation norms.

-- We need to build a new consensus — embracing the Executive and Legislative branches, industry and public, and friends abroad — to promote effective nonproliferation efforts and integrate our nonproliferation and economic goals.

The President reaffirmed U.S. support for a strong, effective nonproliferation regime that enjoys broad multilateral support and employs all of the means at our disposal to advance our objectives.

Key elements of the policy follow.

Fissile Material

The U.S. will undertake a comprehensive approach to the growing accumulation of fissile material from dismantled nuclear weapons and within civil nuclear programs. Under this approach, the U.S. will:

-- Seek to eliminate where possible the accumulation of stockpiles of highly-enriched uranium or plutonium, and to ensure that where these materials already exist they are subject to the highest standards of safety, security, and international accountability.

-- Propose a multilateral convention prohibiting the production of highly-enriched uranium or plutonium for nuclear explosives purposes or outside of international safeguards.

-- Encourage more restrictive regional arrangements to constrain fissile material production in regions of instability and high proliferation risk.

-- Submit U.S. fissile material no longer needed for our deterrent to inspection by the International Atomic Energy Agency.

-- Pursue the purchase of highly-enriched uranium from the former Soviet Union and other countries and its conversion to peaceful use as reactor fuel.

-- Explore means to limit the stockpiling of plutonium from civil nuclear programs, and seek to minimize the civil use of highly-enriched uranium.

-- Initiate a comprehensive review of long-term options for plutonium disposition, taking into account technical, nonproliferation, environmental, budgetary and economic considerations. Russia and other nations with relevant interests and experience will be invited to participate in this study.

The United States does not encourage the civil use of plutonium and, accordingly, does not itself engage in plutonium reprocessing for either nuclear power or nuclear explosive purposes. The United States, however, will maintain its existing commitments regarding the use of plutonium in civil nuclear programs in Western Europe and Japan.

Export Controls

To be truly effective, export controls should be applied uniformly by all suppliers. The United States will harmonize domestic and multilateral controls to the greatest extent possible. At the same time, the need to lead the international community or overriding national security or foreign policy interests may justify unilateral export controls in specific cases. We will review our unilateral dual-use export controls and policies, and eliminate them unless such controls are essential to national security and foreign policy interests.

We will streamline the implementation of U.S. nonproliferation export controls. Our system must be more responsive and efficient, and not inhibit legitimate exports that play a key role in American economic strength while preventing exports that would make a material contribution to the proliferation of weapons of mass destruction and the missiles that deliver them.

Nuclear Proliferation

The U.S. will make every effort to secure the indefinite extension of the Non-Proliferation Treaty in 1995. We will seek to ensure that the International Atomic Energy Agency has the resources needed to implement its vital safeguards responsibilities, and will work to strengthen the IAEA's ability to detect clandestine nuclear activities.

Missile Proliferation

We will maintain our strong support for the Missile Technology Control Regime. We will promote the principles of the MTCR Guidelines as a global missile nonproliferation norm and seek to use the MTCR as a mechanism for taking joint action to combat missile proliferation. We will support prudent expansion of the MTCR's membership to include additional countries that subscribe to international nonproliferation standards, enforce effective export controls and abandon offensive ballistic missile programs. The United States will also promote regional efforts to reduce the demand for missile capabilities.

The United States will continue to oppose missile programs of proliferation concern, and will exercise particular restraint in missile-related cooperation. We will continue to retain a strong presumption of denial against exports to any country of complete space-launch vehicles or major components.

The United States will maintain its general policy of not supporting the development or acquisition of space-launch vehicles in countries outside the MTCR.

For MTCR member countries, we will not encourage new space-launch vehicle programs, which raise questions on both nonproliferation and economic viability grounds. The United States will, however, consider exports of MTCR-controlled items to MTCR member countries for peaceful space launch programs on a case-by-case basis. We will review whether additional constraints or safeguards could reduce the risk of misuse of space launch technology. We will seek adoption by all MTCR partners of policies as vigilant as our own.

Chemical and Biological Weapons

To help deter violations of the Biological Weapons Convention, we will promote new measures to provide increased transparency of activities and facilities that could have biological weapons applications. We call on all nations — including our own — to ratify the Chemical Weapons Convention quickly so that it may enter into force by January 13, 1995. We will work with others to support the international Organization for the Prohibition of Chemical Weapons created by the Convention.

Regional Nonproliferation Initiatives

Nonproliferation will receive greater priority in our diplomacy, and will be taken into account in our relations with countries around the world. We will make special efforts to address the proliferation threat in regions of tension such as the Korean peninsula, the Middle East and South Asia, including efforts to address the underlying motivations for weapons acquisition and to promote regional confidence-building steps.

In Korea, our goal remains a non-nuclear peninsula. We will make every effort to secure North Korea's full compliance with its nonproliferation commitments and effective implementation of the North-South denuclearization agreement.

In parallel with our efforts to obtain a secure, just, and lasting peace in the Middle East, we will promote dialogue and confidence-building steps to create the basis for a Middle East free of weapons of mass destruction. In the Persian Gulf, we will work with other suppliers to contain Iran's nuclear, missile, and CBW ambitions, while preventing reconstruction of Iraq's activities in these areas. In South Asia, we will encourage India and Pakistan to proceed with multilateral discussions of nonproliferation and security issues, with the goal of capping and eventually rolling back their nuclear and missile capabilities.

In developing our overall approach to Latin America and South Africa, we will take account of the significant nonproliferation progress made in these regions in recent years. We will intensify efforts to ensure that the former Soviet Union, Eastern Europe, and China do not contribute to the spread of weapons of mass destruction and missiles.

Military Planning and Doctrine

We will give proliferation a higher profile in our intelligence collection and analysis and defense planning, and ensure that our own force structure and military planning address the potential threat from weapons of mass destruction and missiles around the world.

Conventional Arms Transfers

We will actively seek greater transparency in the area of conventional arms transfers and promote regional confidence-building measures to encourage restraint on such transfers to regions of instability. The U.S. will undertake a comprehensive review of conventional arms transfer policy, taking into account national security, arms control, trade budgetary and economic competitiveness considerations.

DOE F 1325.8
(8-89)
EFG (07-90)

United States Government | Department of Energy

memorandum

DATE: DEC 28 1994

REPLY TO
ATTN OF: EM-37

SUBJECT: Analysis of a Potential New Processing Facility in the Foreign Research Reactor Spent Nuclear Fuel Environmental Impact Statement

TO: Jill E. Lytle
Deputy Assistant Secretary for Waste Management, EM-30

Based on a series of meetings held between staff from EM-4 and EM-30 during early December, I request that you take immediate action to include in the Foreign Research Reactor (FRR) Spent Nuclear Fuel (SNF) Environmental Impact Statement (EIS) an alternative to initiate development work leading to a decision on whether to construct and operate a new SNF processing facility. The following parameters apply to this additional alternative:

- Any new facility would be capable of changing the FRR SNF into a form suitable for geologic disposal, without necessarily separating the fissile materials. A number of alternative processes would ultimately be considered for use in such a facility. Examples of these potential processes should be briefly discussed in the EIS.
- Due to the need for further research and development before the design of such a facility could be selected, the discussion of a new facility will be highly conceptual and programmatic in nature. Further NEPA analysis would be required prior to any decision to construct such a facility.
- Any new facility would be designed to operate safely and to minimize waste volumes, toxicity, and mobility.
- Any new facility would meet or exceed current environmental requirements.
- The alternative should consider construction of a potential new facility at all five of the sites considered for other FRR SNF management activities.
- The discussion should describe the range of quantities of spent fuel that such a facility might be designed to handle (hypothetically, from as little as just the foreign research reactor spent fuel that might be accepted under the FRR SNF EIS to a maximum of all of DOE's spent fuel).
- The design and operation of a new facility would be consistent with U.S. nuclear weapons nonproliferation policies, including the requirements of Presidential Decision Directive 13 regarding reprocessing.

- This alternative considering development potentially leading to a new SNF processing facility is to be in addition to the analysis of chemical separation of the FRR SNF that is already included in the FRR SNF EIS.
- In addition, consideration should be given to utilizing the National Academy of Science to assess the feasibility of using a new facility to produce a waste form that will meet the waste acceptance criteria for a geologic repository.

I recognize that incorporation of this alternative into the FRR SNF EIS at this stage in the development of the EIS will result in approximately a two week delay in the completion of the draft of the EIS. The draft was originally scheduled to be issued for public review and comment by the end of December 1994 and has recently been delayed about two weeks to resolve internal DOE comments. This change will result in a further delay and release of the draft FRR SNF EIS for public review and comment by no earlier than February 1995. This will probably result in a delay in the completion of the final FRR SNF EIS from June 1995 until July 1995. I understand that any delay in the completion of the FRR SNF EIS is likely to raise some objections among the FRR operators. Nevertheless, I consider that it is essential to evaluate this proposed new processing facility and that the small additional delay is acceptable.

Thomas P. Grumbly
Assistant Secretary for
Environmental Management

DEPARTMENT OF STATE
WASHINGTON

October 26, 1992

Dear Mr. Secretary:

During the 1992 International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR) in Denmark, participants voiced very strong concern regarding the apparent reluctance of the Department of Energy to renew the Off Site Fuels Policy, to take back spent research reactor fuel from abroad.

Since 1978, the United States has encouraged countries to convert from the use of high enriched fuel (HEU) to low enriched fuel (LEU). This effort constitutes a key element of U.S. nuclear non-proliferation policy, which has been accepted with some reluctance by other countries, since it entails additional effort and expense on their part. Historically, the Off Site Fuels Policy has been an integral part of the conversion effort, which is perceived by countries as essential to meet reactor operating licensing requirements for disposition of spent fuel and to assure that their research reactor spent fuel is disposed of in a safe and reliable manner.

I fully recognize that renewal of this program will require DOE to resolve difficult and complex budgetary, environmental and technical issues. However, for a variety of reasons, I believe it is essential for DOE to move promptly to renew its policy of taking back foreign research reactor fuel.

We have worked hard for many years to reestablish the position of the United States as a reliable partner in nuclear commerce. We should not forfeit this effort by appearing uncertain about a policy which we have long supported and which is so critical to our non-proliferation objective of eliminating HEU from commercial use.

Clearly, we also do not want to forfeit the significant nuclear non-proliferation gains which have resulted from the RERTR program and our agreement to take back foreign research reactor spent fuel. Limiting the use and location of HEU abroad serves the security interests of both the United States

The Honorable
James D. Watkins,
Secretary of Energy.

and the international community as a whole. Hence, it is particularly disturbing to hear that some countries are considering halting their conversion programs, and even reverting to the use of HEU fuels in the event the United States does not agree to take back U.S.-supplied LEU spent fuel.

Over the past four years, we have maintained a dialogue with DOE concerning the importance of the spent fuel policy. Given the urgent need to resolve this matter, I strongly urge that DOE move quickly to reassure other governments that their spent fuel needs will be fully addressed and that we will continue to honor our commitments to them.

Sincerely,

Larry

Lawrence S. Eagleburger
Acting Secretary

UNITED STATES ARMS CONTROL AND DISARMAMENT AGENCY
Washington, D.C. 20451

THE DIRECTOR

0 7 DEC 1992

MEMORANDUM FOR THE SECRETARY OF ENERGY

SUBJECT: Reducing Foreign Inventories of U.S.-Supplied Highly Enriched Uranium

For many years the United States has encouraged reduced use of highly enriched uranium (HEU) for civil purposes as a key component of U.S. nuclear nonproliferation policy. This effort has met with some success, and the civil use of HEU has diminished, bringing reduced stockpiles and reduced transportation and diversion risks. An important incentive for foreign users of U.S.-supplied HEU to convert their reactors to low enriched uranium (LEU) fuel was the United States' program to take back the spent fuel.

Recent historic political developments have also presented opportunities for further reducing stockpiles of HEU abroad, thereby further promoting our nuclear nonproliferation objectives. We are arranging to purchase 500 metric tons of HEU from Russia for peaceful uses. South Africa has ended its HEU production and has offered to sell its stockpile to the United States.

I believe we shculd consolidate these gains and encourage further reduction of civil HEU use. It is essential to act soon to avoid damaging the longstanding and successful U.S. program that encouraged foreign operators to convert HEU research reactors fueled by the United States to the use of LEU fuel. Without appropriate action, some foreign operators might decide against conversion and others may switch back to HEU fuel. Moreover, new foreign suppliers of HEU may emerge.

In this regard, I have three recommendations:

1. Conclude contractual arrangements with appropriate foreign organizations to take back U.S.-supplied research reactor fuel following any necessary environmental determination.

2. Examine the feasibility of additional incentives that would be helpful or necessary toward ensuring the conversion

of those reactors for which alternative LEU fuels have been identified. A general review of the conversion program may be appropriate in any event in view of the recent amendment to the Atomic Energy Act which severely restricts future HEU licensing. In regard to that legislation, we would also support efforts to reestablish the LEU target development program for production of medical isotopes.

3. Ensure that the United States will make South Africa an attractive offer for its HEU.

I do not underestimate the difficulties posed by these recommendations. However, actions such as these would maintain and strengthen a longstanding and successful U.S. policy of reducing HEU stockpiles abroad -- a policy which will continue to promote global nuclear nonproliferation objectives.

Ronald F. Lehman II

Ronald F. Lehman II

THE DIRECTOR GENERAL

1993-07-01

Dear Madam Secretary,

Since 1978, the United States has encouraged countries to convert the cores of their research and test reactors from the use of highly enriched uranium (HEU) to nuclear fuels of low enriched uranium (LEU). This effort, initiated by President Carter, was an important element of the U. S. non-proliferation policy throughout most of the 1980s and was fully supported through the Reduced Enrichment for Research and Test Reactors (RERTR) programme by the International Atomic Energy Agency. The expiration of the U. S. Department of Energy's Off-Site Fuels Policy (the Policy) in 1988 has led to a crisis for the operators of research reactors in many countries where the laws are such that continuation of licensing and/or purchase of new nuclear fuels is contingent upon a resolution of spent fuel management problems. This situation is exacerbated for many reactor operators who complied with the wishes of the U. S. and converted their cores to LEU. They now have interim storage pools filled with irradiated HEU fuels and are trying to cope with a greater throughput of LEU fuels. The anticipated announcement that the U.S. DOE will renew the Policy and in due course begin the take back of research reactor fuels of U.S. origin from around the world will be very much welcomed by the Agency and many of its Member States.

However, because of the problems of spent fuel management facing the operators of many research reactors the Agency urges the earliest implementation of the Policy renewal. Some of these research facilities are the only sources of radioisotope production for medical uses in the countries in question, but face imminent closure unless they can resolve their problems of spent fuel management quickly. The Agency has initiated programmes to advise them, but the real solution for most of them is to return their irradiated research reactor fuels of U. S. origin. It is understood that the renewal of the Policy will require the solution of difficult and complex budgetary, environmental, transportation, legal and technical issues. Nevertheless, the Agency is confident that when the resources of the U. S. DOE are brought to bear on these problems that they will be resolved as soon as possible.

The Honourable Hazel O'Leary
Secretary of Energy
Washington, DC 20585
United States of America

Limiting the use and location of HEU fuels throughout the world remains a valuable objective and will serve the security interests of all nations. The Agency stands ready to help in any way it can consistent with its mandate and budgetary constraints.

Yours sincerely,

Hans Blix

THE SECRETARY OF STATE

WASHINGTON

July 2, 1993

Dear Madam Secretary:

I am writing to urge your personal support for renewal by the Department of Energy of the Off Site Fuels Policy for the acceptance of spent research reactor fuel from abroad.

The Department of State has strongly supported this policy because of its importance in gaining foreign cooperation in converting reactors from highly enriched uranium (HEU) to low enriched (LEU) fuel under the aegis of the Reduced Enrichment in Research and Test Reactors (RERTR) Program.

We recall Secretary Watkins confirmed in 1992 that the Department of Energy proposed to renew the Off Site Fuels Policy, but with the caveat that meeting the requirements of the National Environmental Policy Act (NEPA) could take as long as 2 to 3 years. We are concerned, however, about reports of substantial delays in the amendment of the existing Environmental Assessment, an essential early step in the NEPA process.

Foreign research reactor operators are reportedly highly concerned about a perceived change in DOE policy and have threatened to withdraw from further RERTR cooperation and to seek resumption of HEU supply from sources such as Russia.

A breakdown of the international consensus on conversion of research and test reactors to LEU and a return to an HEU fuel economy would undermine 15 years of intensive U.S. non-proliferation efforts on this matter and substantially reduce the ability of the U.S. to influence nuclear policy in bilateral and international fora.

In light of current developments, I urge your support for early reaffirmation by DOE to other governments of our continued commitment as a reliable supplier to fully address their spent fuel needs.

Sincerely,

Warren Christopher

Warren Christopher

The Honorable
Hazel R. O'Leary,
Secretary of Energy.

The Secretary of Energy
Washington, DC 20585

July 13, 1993

The Honorable Warren Christopher
Secretary of State
Washington, D.C. 20520

Dear Mr. Secretary:

This is in response to your letter dated July 2, 1993, urging my support for renewal of the Department of Energy's policy for the acceptance of spent research reactor fuel from abroad.

The Department of Energy remains committed to the Reduced Enrichment for Research and Test Reactors (RERTR) program, and to the proposal to establish a policy for the return of U.S. origin spent fuel from foreign research reactors. In response to your letter, and other inquiries we have received on this subject, we have taken a hard look at how we can expedite actions in these areas. We have decided on a three-tiered approach, as follows:

1. For any foreign research reactor spent fuel returns for which we can mutually agree that a bona fide emergency exists, the Department of Energy will join with you in consulting with the Council on Environmental Quality on the implementation of alternative arrangements for compliance with environmental review requirements pursuant to the emergency provisions of the Council on Environmental Quality"s regulations implementing the National Environmental Policy Act (40 CFR 1506.11).

2. In order to be able to respond to any near-term situation in which the expiration of the Department's acceptance of foreign research reactor spent fuel may threaten the Reduced Enrichment for Research and Test Reactors Program, the Department has begun an expeditious Environmental Assessment of the proposed return of sufficient spent fuel to eliminate that threat. It is proposed that any near-term spent fuel returns would be conducted under the terms and conditions of the enclosed proposed policy and be limited to approximately 550 spent fuel elements which can be stored in existing DOE capacity. This Environmental Assessment is scheduled to be completed by September 1993, and, if appropriate, a proposed Finding of No Significant Impact will be issued for public review by no later than September 30, 1993. Our goal is to complete the National Environmental Policy Act review process of this proposed limited foreign research reactor spent fuel acceptance by the end of this calendar year.

3. For the longer term, the Department will undertake preparation of an Environmental Impact Statement that addresses the proposed return of all U.S. origin foreign research reactor spent fuel, as specified in the enclosed proposed policy. A notice of intent for preparation of this Environmental Impact Statement is in preparation and should be issued in August 1993. The Department intends to issue the draft of the Environmental Impact Statement for public review by no later than the

end of December 1994, and the final Environmental Impact Statement by the end of June 1995.

We cannot continue to address this issue in a business as usual manner. The actions outlined above reflect our determination to move forward promptly and our acknowledgement of the need for a new definition of national security - one that includes both nonproliferation and environmental concerns. To provide added emphasis to the urgency of this effort, the Department requests that the Department of State participate as a cooperating agency in preparation of this environmental documentation.

In conclusion, the Department is committed to work with you and representatives of the Council on Environmental Quality at any time that you consider an emergency situation may be developing. In the meantime, we are proceeding as expeditiously as possible on the actions outlined above.

Sincerely,

Hazel R. O'Leary

Hazel R. O'Leary

Enclosure

DRAFT

Proposed Foreign Research Reactor Spent Nuclear Fuels Acceptance Policy

13 July 1993

<u>PURPOSE</u> - This proposed Department of Energy policy would support United States nonproliferation policy, including one of its key elements, the Reduced Enrichment Research and Test Reactors Program. It would provide opportunities and incentives for research reactor operators in foreign countries holding United States origin spent nuclear fuel containing highly enriched uranium to return that spent nuclear fuel to the United States for storage and eventual geologic disposal. This proposed policy is intended to support the United States nonproliferation objective of eliminating United States origin highly enriched uranium from research reactor use. It is also consistent with Section 903(a) of the Energy Policy Act of 1992, which places further restrictions on the export of highly enriched uranium from the United States. This proposed policy would provide incentives to encourage and assist developing countries (defined below) in returning their United States origin highly enriched uranium research reactor spent nuclear fuel to the United States for storage and disposal. For developed countries, the policy would allow return of United States origin research reactor spent nuclear fuel to the United States for storage and disposal on a full-cost-recovery basis.

<u>PROPOSED POLICY</u> - The United States proposes to adopt a policy under which:

1. <u>For developing countries</u> (i.e., those eligible for assistance under the United Nations Assistance Program), the United States would offer to accept United States origin research reactor spent nuclear fuel containing highly enriched uranium for storage and disposal in the United States. The United States would reimburse the developing country for costs incurred in transportation of the spent nuclear fuel from the developing country to a receipt facility in the United States. Upon acceptance of the spent nuclear fuel in the United States, the United States would assume all responsibility for the spent nuclear fuel, including storage of the spent nuclear fuel in the United States, any preparation of the spent nuclear fuel for disposal, all transportation in the United States subsequent to spent nuclear fuel acceptance, and ultimate geologic disposal of the spent nuclear fuel in the United States.

2. <u>For developed countries</u>, the United States would offer to accept all United States origin research reactor spent nuclear fuel containing highly enriched uranium for storage, preparation for disposal, and eventual geologic disposal in the United States. Such acceptance would be conducted on a full-cost-recovery basis, with the developed country responsible for transportation of the spent nuclear fuel to a designated receipt facility in the United States and paying the United States the full cost of all storage, all transportation within the United States subsequent to spent nuclear fuel acceptance, disposal preparation, and ultimate geologic disposal.

3. <u>To encourage the conversion of foreign research reactors currently using United States origin highly enriched uranium fuels to low enriched uranium fuels</u>, the United States would offer to accept for storage and ultimate disposal certain United States origin low enriched uranium research reactor spent nuclear fuel. Specifically, low enriched uranium research reactor spent nuclear fuel of United States origin would be accepted for a ten year period following implementation of this policy from reactors that have already converted, or that were constructed to use and operate with low enriched uranium fuels. United States origin low enriched uranium research reactor spent nuclear fuel exported to research reactors that convert within five years of the effective date of this policy would also be accepted for a ten-year period following their initial order for low enriched uranium fuel.

   The acceptance of low enriched uranium research reactor spent nuclear fuel from developed and developing countries would be conducted on the same terms as stated in 1 and 2 above for highly enriched uranium research reactor spent nuclear fuel.

<u>CONDITIONS</u>

1. This proposed policy would apply only to receipt of spent research reactor nuclear fuel of United States origin.

2. Ownership of the spent nuclear fuel would be transferred to the United States upon acceptance of the spent nuclear fuel by the United States at a designated receipt and inspection facility in the United States.

3. All transportation within a developing country and to the United States receipt facility would be the responsibility of the developing country, but would be paid for by the United States (subject to United States approval of the transportation arrangements and costs).

   All transportation within a developed country and to the United States receipt facility would be the responsibility of, and would be paid for by, the developed country.

4. Criteria concerning the required condition of the spent nuclear fuel would be published by the United States as part of the announcement of this policy, to clarify conditions for acceptance of the spent nuclear fuel. In general terms, all spent nuclear fuel to be accepted by the United States would be required to be either intact and free of defects or canned to ensure the ability to safely contain and manage the spent nuclear fuel.

5. For developed countries, the fee to be paid to achieve full cost recovery would be established by the Department prior to entering into the agreements to accept the spent nuclear fuel. This fee would be based on <u>estimates</u> of the cost of the storage and disposal activities that would be required. The fee

schedule would be updated annually to account for items such as inflation, and experience with the program.

TERMINATION - This policy of accepting low enriched uranium research reactor spent nuclear fuel would expire ten years after the effective date of this policy (or ten years following placement of an order for low enriched uranium research reactor fuel to replace highly enriched uranium research reactor fuel, if such an order is placed within five years of the effective date of this policy). Therefore, countries and research reactor operators that plan to take advantage of this policy for spent nuclear fuel containing low enriched uranium should begin planning for their own national or regional means of storage and disposal of low enriched uranium research reactor spent nuclear fuel for use following termination of this policy.

The proposed policy for accepting research reactor spent nuclear fuel containing highly enriched uranium of United States origin would encourage all countries to return this United States origin research reactor spent nuclear fuel as soon as possible.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 03 1991

Mr. John J. Easton, Jr.
Assistant Secretary of Energy
International Affairs and Energy Emergencies
U.S. Department of Energy
Washington, DC 20585

Dear Mr. Easton:

I am responding to your letter of May 31, 1991, requesting the comments of the Nuclear Regulatory Commission on issues related to the Department of Energy's consideration of renewing the Off-Site Fuels Policy.

The NRC staff believes that it is in the best interest of the United States to allow spent U.S.-origin high enriched uranium (HEU) fuel from domestic and foreign research reactors to be returned to DOE for processing and storage. Such a take-back policy reduces certain safeguards, physical security and safety concerns associated with the indefinite, long-term storage of irradiated HEU fuel in diverse locations. It would, of course, also alleviate the serious lack of spent fuel storage capacity being experienced by several research facilities, including ones in Japan and several European countries. In this regard, however, we assume that in implementing a resumption of the DOE policy to accept spent HEU fuel, the U.S. would not diminish its pressure on foreign countries to continue their best efforts to convert remaining HEU-fueled research reactors to low enriched uranium (LEU) fuel.

In the same vein, it would appear useful for DOE also to extend, beyond the expiration date of December 31, 1992, its offer to take back spent U.S.-origin LEU research reactor fuel from domestic and foreign users. DOE's current examination of the Off-Site Fuels Policy will no doubt address the question of whether or not this commitment is essential to U.S. efforts to minimize the use of HEU fuel in research reactors abroad. Your analysis of this and other aspects of the policy will be of great interest to NRC and can be expected to influence our future export licensing activities.

I trust that these general comments are useful.

Sincerely,

James M. Taylor
Executive Director for Operations

THE SECRETARY OF STATE

WASHINGTON

December 12, 1995

Dear Madam Secretary:

As we move to the final stages of preparing the Environmental Impact Statement on a Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel (the EIS), I want to reaffirm the critical need for implementing this policy. The spent fuel acceptance policy which the EIS supports is central to our goal of preventing the spread of nuclear weapons -- and therefore to a major national security objective of this Administration.

One of the key elements of the President's nonproliferation policy has been to minimize and eventually to eliminate the use of high enriched uranium (HEU) in civil world commerce. I greatly appreciate the efforts that you personally have made to reinvigorate the Reduced Enrichment for Research and Test Reactors (RERTR) program which is designed to convert research reactors around the world from using HEU to LEU fuel.

As you know, the willingness of research reactor operators to support this vital program depends on our willingness to assist them with disposition of the spent fuel produced from nuclear materials which the United States originally supplied. Failure to implement this policy successfully would deal a crippling blow to our efforts to minimize the commercial use of HEU.

I hope that you will proceed soon to publish the final EIS and to begin implementing the policy using available practical and appropriate means to ensure the expeditious implementation of the program.

You and your staff are to be commended for the outstanding effort made in preparing the EIS. The Department of State stands ready to cooperate with you in whatever way we can.

Sincerely,

Warren Christopher

Warren Christopher

The Honorable
Hazel R. O'Leary,
Secretary of Energy

DOE/EIS-0218F
February 1996

Volume 2

FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix H
General Provisions of Transportation Planning for the Shipments of Foreign Research Reactor Spent Nuclear Fuel

United States Department of Energy
Assistant Secretary for Environmental Management
Washington, DC 20585

Table of Contents

List of Figures

Appendix H
General Provisions of Transportation Planning for the Shipments of Foreign Research Reactor Spent Nuclear Fuel

H.1 Overview

This appendix is prepared to provide a description of the transportation, emergency response, security, and communications planning that would typically occur prior to any acceptance of foreign research reactor spent nuclear fuel under Management Alternative 1 or Management Alternative 3. Appendix H expands upon Sections 2.0 (Proposed Action and Alternatives) and 2.7 (Characteristics of Emergency Management and Response) in this Environmental Impact Statement (EIS). The information in this appendix is based on U.S. Department of Energy (DOE), State, Tribal, and local authorities' composite experience in planning for several successful radioactive material shipping campaigns, such as Urgent Relief Acceptance of Foreign Research Reactor Spent Nuclear Fuel and Cesium 137 Capsule Return Program shipments. This appendix is not meant to be all-inclusive as each shipping campaign differs slightly because of the material being shipped, the transportation mode, and the level of involvement of State, Tribal, and local governments.

The implementation of Management Alternative 1 or Management Alternative 3 would involve an ongoing and interactive planning process between DOE, States, Tribes, local authorities, other Federal agencies, the shipper of record/shipper's agent, and the carrier. In past shipping campaigns, these participants have developed an overall "Transportation Plan." This Plan is a blueprint for transportation, emergency response, security, and communications operations that would take place prior to, during, and after the completion of a shipment. Agreements between the various parties are detailed in the Plan. What follows is a description of the main elements of such a Plan. Additional information is available in the *DOE Program Manager's Guide to Transportation Planning* (DOE, 1995).

H.2 Transportation Considerations

Transportation planning integrates a wide range of expertise and requirements, including program management, material handling and packaging, transportation operations (traffic management), key governmental involvement, public information, environmental safety and health, and emergency preparedness. Planning would be clarified in a Transportation Plan that would document the planned logistics for foreign research reactor spent nuclear fuel shipments. The focus of this Plan would be operational; e.g., the handling, packaging, and transport of the foreign research reactor spent nuclear fuel shipment through sequential steps resulting in safe transport of this material to a management site. The plan would include organizational responsibilities of DOE, foreign research reactor operator/shipper of record, corridor jurisdictions and other Federal agencies. It would contain information on shipment schedules, the port(s) of entry, the mode of transport from port(s) of entry to a selected management site, an illustration of the shipment route, emergency plans and contacts, and communications strategies. The plan would include graphic representation of schedules of requirements for pre-, during, and post-shipment activities showing number of days to prepare, load, ship, unload and return the empty cask.

H.2.1 Organizational Roles and Responsibilities

DOE Headquarters: DOE Headquarters sets overall policy for the spent nuclear fuel program and for transportation, resolves policy questions, issues guidance, and provides information for use in transportation activities. DOE Headquarters provides a management team that offers general guidance and technical assistance to the field office implementing the program activities. As the responsible government agency for this program, DOE would ensure overall program coordination with involved organizations and agencies as outlined in this appendix. DOE is committed to providing corridor jurisdictions with technical assistance to help prepare for any shipments, and to supporting the management sites in implementing this program if foreign research reactor spent nuclear fuel is accepted into the United States. Prior to the implementation of a foreign research reactor spent nuclear fuel acceptance policy, DOE would coordinate with corridor jurisdiction authorities to resolve issues related to transport, emergency response, security, and communications management. DOE Headquarters would ensure, through contractual agreement, that the foreign research reactor operators comply with DOE's transportation, emergency response, security, and communications provisions, which exceed U.S. Nuclear Regulatory Commission (NRC) and/or U.S. Department of Transportation regulatory requirements.

Management Sites: The management site would be responsible for overall program management for shipments to the storage facility and transfer of spent nuclear fuel elements from the foreign research reactor to the management site. The management site would require shipping activities to be in compliance with applicable regulations and considerations. When necessary based on shipping experience and reactor operator capabilities, DOE would have a team of specialists from the management site travel to foreign research reactor facilities to verify that spent nuclear fuel and shipping arrangements meet DOE's transportation acceptance requirements. The management site would coordinate all communications activities with DOE Headquarters, other Federal agencies, and corridor jurisdiction authorities. Unloading and cross-site movement to a temporary storage facility would be conducted in accordance with management site operating procedures for acceptance and unloading of spent nuclear fuel.

Reactor Operators: The reactor operator would either perform the duties of the "Shipper of Record", as specified below, or contract with a shipping firm to act as the "Shipper of Record."

Shipper of Record: The Shipper of Record would submit a composite transportation physical security plan to the NRC, for domestic transport, which would contain a U.S. Department of Transportation-approved or State-designated route for highway or rail shipments. At DOE's request, the Shipper of Record would also develop its transportation planning document, which would address considerations such as vessel selection, pre-notifications, import authorizations, port arrangements, U.S. customs clearance, carrier arrangements, schedule, and emergency response. The reactor operator and/or the Shipper of Record are often referred to as the licensee. The licensee refers to the license that has been granted by the NRC to handle spent nuclear fuel.

Other Federal Agencies: Other Federal agencies would cooperate with DOE to ensure safe transportation of the foreign research reactor spent nuclear fuel. These agencies include:

> ***Department of Transportation:*** The U.S. Department of Transportation would perform rail track inspections; ensure that each shipment was in compliance with the regulations for the transport of spent nuclear fuel; and serve as the U.S. Competent Authority in the review of foreign transportation packaging certificates. The U.S. Department of Transportation also maintains a list of State-designated alternative highway routes.

NRC: The NRC would be responsible for review and approval of the composite transportation physical security plan, which would include the routes submitted by the foreign research reactor operators or the shipper of record. The NRC could perform radiological monitoring of the transportation packages at the port.

Department of Defense: If a military port-of-entry were utilized, organizational responsibilities would be defined and agreed upon in advance between DOE and the U.S. Department of Defense. Interagency Agreements would be established between DOE and the appropriate U.S. Department of Defense element to define the provisions under which DOE would be allowed to ship through the U.S. Department of Defense port. Security and safeguard measures would comply with military requirements in addition to NRC requirements.

Corridor Jurisdictions: State, Tribal, and local authorities have primary responsibility for the health and welfare of their citizens. Most State, Tribal, and local emergency preparedness organizations have, as a minimum, the basic capabilities to respond to a transportation emergency. In addition, the following authorities could be exercised:

- Corridor jurisdictions could exercise responsibility for vehicle and equipment inspections.
- Corridor jurisdictions would be responsible for notifying DOE of any road, rail or weather conditions that could affect a shipment crossing their jurisdiction.
- Corridor jurisdictions would designate a central point of contact.
- Corridor jurisdictions would interact with local officials on information and emergency planning activities.
- If highway transport were utilized, DOE, corridor jurisdictions, and the carrier would jointly establish policies regarding bad weather/road conditions and safe parking procedures.

Carrier: The carrier would be responsible for safely transporting the spent nuclear fuel packages and transport containers from the port of entry to the management site. If a shipment were to be completed by rail, this responsibility would include ensuring that rail lines and equipment were properly inspected and in good operating order, following NRC approval of the composite transportation physical security plan, and coordinating with the corridor jurisdictions to arrange inspections as required. If shipment were by highway, this responsibility could include obtaining required State permits, using the designated highway route-controlled quantity route, and arranging for vehicle inspections, as required. The carrier would be required to have a transport plan addressing considerations such as emergency recovery, transportation regulation compliance, two-way communication with rail/truck operators, and subcontractor emergency response plans.

H.2.2 Advance Information

Prior to the transport of spent nuclear fuel within or through a State, the Shipper of Record would notify the Governor or the Governor's designee in writing.

H.2.3 Tracking of Shipment

For shipments of foreign research reactor spent nuclear fuel, DOE policy would require that a satellite tracking/communication system be used.

H.2.4 Implementation of United States/International Atomic Energy Agency Agreement

The International Atomic Energy Agency is an agency of the United Nations headquartered in Vienna, Austria. The International Atomic Energy Agency establishes standards for radioactive materials transport. These model regulations may be adopted by individual nations. The emphasis of the International Atomic Energy Agency model regulations is on package integrity. The NRC and the U.S. Department of Transportation both periodically review and revise their regulations to bring them into general accord with the International Atomic Energy Agency regulations.

H.2.5 Packaging Description

The packaging for transporting spent nuclear fuel is designed to provide containment of its contents as required by international and Federal regulation. Type B packages, used to transport spent nuclear fuel, are designed to protect and retain their contents in both normal and severe accident conditions. Foreign-licensed Type B package designs are reviewed and approved for acceptance by the U.S. Department of Transportation. Figure H-1 displays a transportation cask typically used to transport spent nuclear fuel.

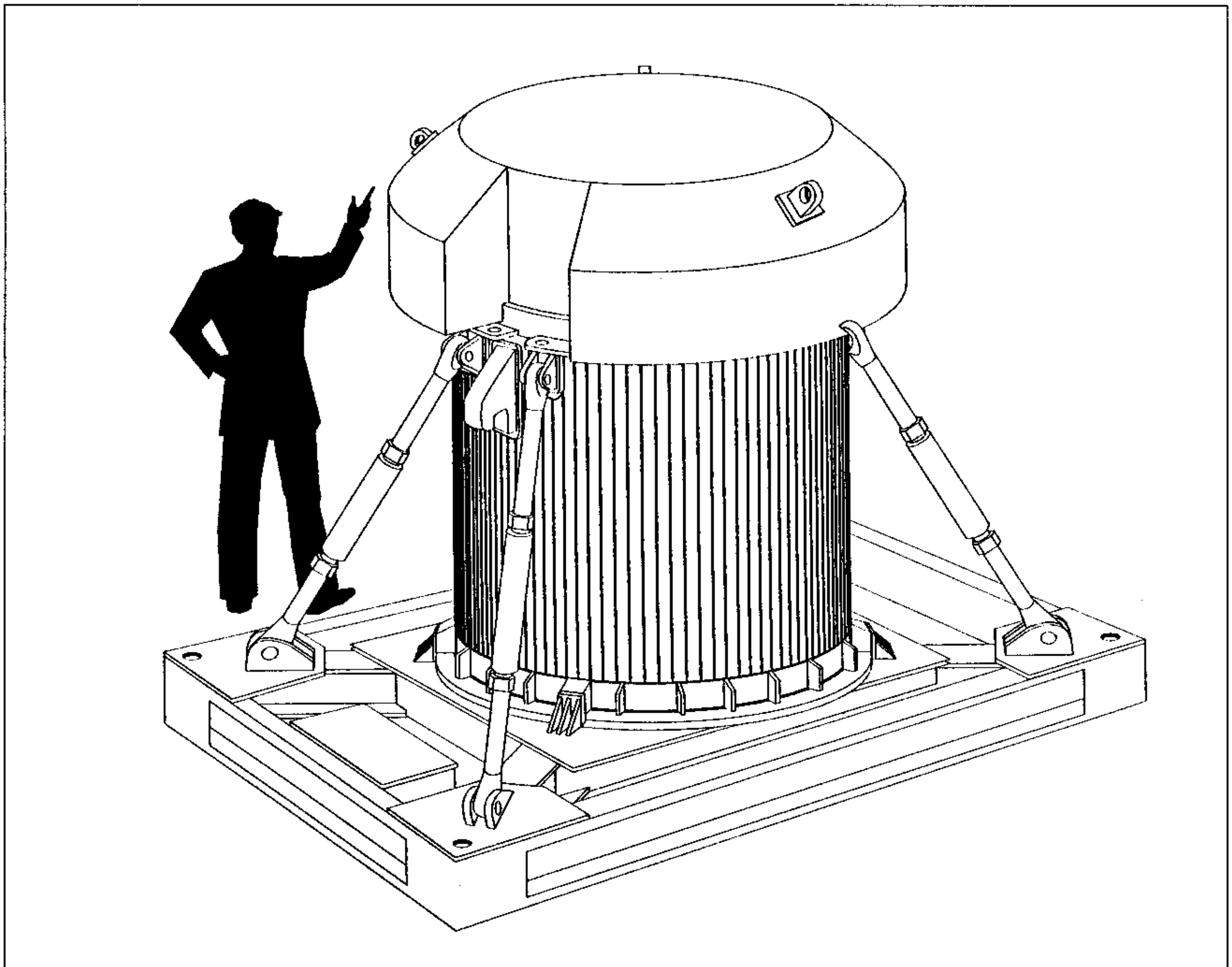


Figure H-1 Typical Spent Nuclear Fuel Transportation Cask

H.3 Emergency Preparedness Considerations

Emergency preparedness for transport of spent nuclear fuel is a vital part of the transportation planning process. Corridor jurisdictions having authority over areas through which these shipments would pass have primary responsibility for protecting the public and the environment and for establishing incident command in the unlikely event that an emergency should occur involving the shipments. DOE would work with State, Tribal, and local authorities to complement existing emergency preparedness capabilities. The carrier for these shipments would be responsible for providing emergency response assistance and recovery/restoration actions, if required. DOE would provide technical operations advice and radiological monitoring assistance to civil authorities and carriers of these shipments, when requested.

H.3.1 Emergency Preparedness

Corridor jurisdiction authorities have primary responsibility for the health and welfare of their citizens. Most States maintain specialized teams capable of responding to hazardous materials incidents. Through the capabilities these teams currently possess for dealing with potential accidents involving other hazardous materials (i.e., hazardous chemicals), they should already have the capability to deal with most plausible accidents involving spent nuclear fuel. Nevertheless, to assist in planning and preparedness for an unlikely but theoretically possible transportation emergency involving any foreign research reactor spent nuclear fuel shipments, DOE would offer a variety of emergency response resources and information to complement existing emergency preparedness programs, and also maintains a comprehensive emergency management system, particularly for radiological emergencies. The emergency management system includes training courses, Regional Coordinating Offices, and DOE Radiological Assistance Program teams.

***Corridor Jurisdictions Hazardous Materials or Radiological Response Teams*:** Most corridor jurisdictions maintain specialized hazardous materials response teams that could be activated to provide technical assistance and mitigation during emergencies. State teams are activated at the request of an Incident Commander or other appropriate State or local authority.

***Carrier Emergency Response Assistance*:** The carrier would provide technical response assistance to corridor jurisdiction responders as required by event scene conditions.

DOE Radiological Assistance Program: DOE's Radiological Assistance Program teams are administered by eight Regional Coordinating Offices. Each Regional Coordinating Office has access to radiological monitoring and assessment capabilities to provide assistance in radiological emergencies. Additional DOE technical experts are available to provide advice on material characteristics and mitigation, packaging and its tie-downs, and radiological monitoring and assessment requirements.

H.3.2 Notifications and Communications

In the unlikely event of a transportation incident involving foreign research reactor spent nuclear fuel, the carrier operator would notify the appropriate State and local authorities, the carrier dispatch center, and the management site communication center. The management site would also inform the appropriate corridor jurisdiction authorities. In any case, State, Tribal, and local authorities would be tied into the transportation plan's communication network arranged for the shipment.

The management site would serve as the designated communication center. DOE policy for transporting foreign research reactor spent nuclear fuel requires that a satellite communication system be used. The carrier may also provide its own transportation tracking system. Further details for the notification, communications, and other responsibilities of the communication center would be outlined in the Shipper of Record's transportation plan.

H.3.3 Emergency Response

If an accident requiring emergency response were to occur, the following emergency-related roles and responsibilities would be provided:

***Carrier Response*:** If an accident were to occur, the carrier operator and/or escort would notify local emergency response personnel as predetermined in both the DOE and Shipper of Record's Transportation Plans. They would also undertake first aid actions and initial incident scene control, provide assistance to first responders, and other emergency actions as defined in the carrier's emergency plan. The carrier operator would provide technical response assistance to State, Tribal, and local responders as required by event scene conditions.

***First Responders*:** Local emergency response personnel would respond to the incident scene when notified by a predetermined notification network initiated by the "Initial Responder." Their first action would be the evaluation of the accident scene (with assistance from the "Initial Responder") for the presence of radiological or other hazards. The response personnel would then act to reduce the hazard and control the event scene. The coordination of the accident scene would typically be under the Incident Command System utilized by local fire departments.

Responders would have information sources available to them in the form of the Department of Transportation *Emergency Response Guidebook* and by available emergency response information accompanying the shipping papers and normally available to responders in the vehicle, or accessible via the satellite communications and tracking system. Response to other hazards identified at the accident scene would be guided by information also contained in the Emergency Response Guidebook or other appropriate protective measures and response guidelines. In all cases, the Incident Commander would be a corridor jurisdiction authority. If States and local responders have additional procedures that provide more specific guidance, responders would follow those procedures.

Local organizations typically involved in first on-scene response include:

- Law enforcement,
- Emergency medical services,
- Cognizant transportation department,
- Hazardous materials team, and
- Fire services.

***Responder Support*:** Response organizations would arrive at the incident scene to support and assist the initial and first on-scene responders as requested by the "First Responders." The Incident Commander would coordinate the actions of these trained personnel coming from agencies within or outside the initial response jurisdiction. Requests for resource augmentation could be performed through local mutual aid associations as part of a response support network. Utilization of response resources would be based upon communication and coordination with State, Tribal, and local agencies, and shipper/carrier response personnel. For incidents that exceed the capabilities of local, State, or Tribal government, Federal assistance could be requested from DOE. DOE could provide Radiological Assistance Program teams that include personnel and instruments for radiological monitoring, provide medical advice, and request assistance from other Federal agencies.

"Responder Support" assistance could involve traffic and/or access control, support of incident mitigation activities, appropriate notification (hospitals, mutual-aid emergency management system), technical assistance for assessment of health risks, and coordination of emergency health services or technical assistance for assessment of environmental risks and coordination of emergency planning for cleanup and recovery as defined in applicable emergency response plans. Organizations typically involved in responder support include:

- Fire services,
- Law enforcement,
- Cognizant transportation department,
- Hazardous materials team,
- Health protection oversight,
- Environmental oversight, and
- DOE Radiological Assistance Program team.

***Package Recovery Actions*:** If package recovery, repositioning, or placement on another vehicle were required, provisions for necessary service would be prearranged. The carrier would have primary responsibility for package and transporter recovery operations. These activities would not begin until the emergency phase of any accident was terminated, following a decision that no radiological or other hazard was present. Recovery planning is initiated prior to termination of an emergency. DOE would assist the carrier in recovery operations where appropriate. Specific procedures for shipping cask recovery would be included in the shipper of record's agent/carrier's transportation plan. Corridor jurisdictions could exercise highway vehicle inspection authority before permitting the recovery vehicle to continue to a management site. Organizations typically involved in recovery actions include:

- Shipper's representative
- Carrier representatives,
- Management sites representative,
- DOE contractor representatives, and
- State, Tribal, and local authorities.

Cleanup and Incident Scene Restoration: In over forty years of experience with spent nuclear fuel shipments, there has never been an incident in which a spent nuclear fuel transportation cask has released any of its contents, even as a result of an accident. In the unlikely event that there was a release of radioactive material, DOE would be ready to provide whatever assistance was needed to respond to the situation. On the other hand, cleanup of nonradioactive aspects of an accident and scene restoration are a part of any accident response. The carrier would have primary responsibility for cleanup and site restoration following an emergency and would provide the necessary resources. Cleanup planning could be initiated prior to termination of an emergency. Standards for such actions would be established by regulation and by authorities in the affected jurisdiction(s). Organizations typically involved in cleanup/restoration oversight actions include:

- Carrier representatives,
- Carrier cleanup/restoration representatives,
- Management site representative,
- DOE contractor representatives,
- Federal environmental oversight, and
- State, Tribal, and local authorities.

Carriers are financially responsible for accident response. The carrier is responsible for maintaining $5 million of insurance to cover costs incurred from an accident. Cost incurred by local first responders (firefighters, police, etc.,) to an incident scene are part of the carrier's financial responsibility. Further, the Price-Anderson Amendment Act ensures coverage of cost incurred beyond the $5 million carrier limit for spent nuclear fuel accidents. The Price-Anderson Act was partly established to ensure that funds are available to compensate the public for personal injury and property damage caused by the release of radioactivity (NRC, 1988). Such coverage would only take effect if the Price-Anderson Act conditions are met such as the resultant damage from the release of radioactivity during the accident exceeded the liability protection of the carrier.

H.4 Security Considerations

Domestic transportation of the foreign research reactor spent nuclear fuel would be regulated by the U.S. Department of Transportation and NRC. The objectives of the security measures employed during foreign research reactor spent nuclear fuel shipments would be to minimize the possibilities for radiological sabotage of spent nuclear fuel shipments and facilitate the location and recovery of spent nuclear fuel shipments that could have come under the control of unauthorized persons. The elements of the security systems to be considered when developing a transportation plan and are briefly summarized below.

Security Packaging: Type B packages are used to transport spent nuclear fuel and would have a seal, which is not readily breakable, and which, while intact, would be evidence that the package has not been opened by unauthorized persons.

General Security Concept: The physical security system would include procedures for coping with circumstances that threaten deliberate damage to a spent nuclear fuel shipment and with other safeguards emergencies; instructions for each escort would be developed. These procedures would include detection of the abnormal presence of unauthorized persons, vehicles, or vessels in the vicinity of a spent nuclear fuel shipment or upon detection of a deliberately induced situation that would have the potential for damaging a spent nuclear fuel shipment; monitoring the progress of the spent nuclear fuel shipment to notify the appropriate agencies in the event a safeguards emergency should arise; and maintaining a written log by the escorts and communications center personnel for each spent nuclear fuel shipment. Arrangements would be made with local law enforcement agencies along the routes of road and rail shipments, and at U.S. ports where vessels carrying spent nuclear fuel shipments were docked, for their response to a security event or a call for assistance. Advance NRC approval of the U.S. Department of Transportation or State-designated alternative routes used for road and rail shipments of spent nuclear fuel, and of any U.S. ports where vessels carrying spent nuclear fuel shipments were scheduled to stop would be required. Shipments would be planned so that scheduled intermediate stops would be avoided to the extent practicable; at least one escort would maintain visual surveillance of the shipment during periods when the shipment vehicle was stopped, or the shipment vessel was docked. Shipment escorts would make calls to the communications center at least every two hours to advise of the status of the shipment for road and rail shipments, and for sea shipments while shipment vessels were docked at U.S. ports. These escorts (other than members of local law enforcement agencies, or ship's officers serving as unarmed escorts) would have successfully completed required training. In addition to NRC licensee requirements, DOE and the licensee could develop extra-regulatory guidelines, as necessary. These guidelines would be established in the overall transportation plan.

***Shipment by Sea*:** The management site would provide a representative at the shipment point of origin to observe the preparation and loading of the material. NRC could also send a representative to the point-of-origin to inspect transport packages and conduct radiological surveys prior to departure.

Advance NRC approval of the routes used for road and rail shipments of spent nuclear fuel, and of any U.S. ports where vessels carrying spent nuclear fuel shipments were scheduled to stop is required. The local law enforcement agencies, at U.S. ports where the ship was docked, would be contacted and arrangements would be made for their response in the event of an emergency situation concerning the spent nuclear fuel.

A shipment vessel, while docked at a U.S. port within a heavily populated area, would be protected by two armed escorts stationed on board the shipment vessel, or stationed on the dock at a location that would permit observation of the shipment vessel; or a member of a local law enforcement agency equipped with normal local law enforcement agency radio communications, who would be stationed on board the shipment vessel, or on the dock at a location that would permit observation of the shipment vessel.

A shipment vessel, while within U.S. territorial waters or while docked at a U.S. port not within a heavily populated area, would be accompanied by an escort (e.g., an officer of the shipment vessel's crew), who would assure that the shipment were unloaded only as authorized by the Shipper of Record.

The escorts would have the capability of communicating with the communications center and local law enforcement agencies through the use of a radiotelephone, or other NRC-approved equivalent means of two-way voice communications.

If a military port were used, applicable requirements would be established in advance among the U.S. Department of Defense, Shipper of Record, DOE, and NRC to provide the appropriate level of security, which would meet or exceed the security provided at commercial ports by the provisions discussed above.

***Shipments by Highway*:** For any shipment of spent nuclear fuel, the Shipper of Record would provide physical protection in compliance with a plan established under requirements prescribed by the NRC or equivalent requirements approved by the Association Administrator for Hazardous Materials Safety. An NRC regulated shipment requires advance approval of the Department of Transportation or State-designated agencies for alternative routes that would be used for highway shipments of spent nuclear fuel. Arrangements would be made with local law enforcement agencies along the transportation route for response to security events or calls for assistance while the shipment was within their jurisdiction. Highway shipments would be accompanied by escorts. Escorts are required by regulations to ride in the shipment vehicle (truck cab) or in separate vehicles maintaining visual sight at all times. Additionally, escorts would have the capability to communicate with their communications center, each other, and the local law enforcement agency. The local law enforcement agency would be able to direct a prompt response to counter a threat to the spent nuclear fuel. The transport vehicle would be equipped with a feature approved by the NRC, which would allow the driver or on-board escort to immobilize the cab or cargo-carrying portion of the vehicle should an attempt be made to seize control of the vehicle. Both the driver and the on-board escort would have appropriate training.

Shipment by Rail: The NRC requires advance approval of the routes used for rail shipments of spent nuclear fuel. A rail shipment car within a heavily populated area would be accompanied by two armed escorts (who are often members of the local law enforcement agency). At least one escort would be stationed at a location on the train that would permit observation of the shipment car while in motion.

A rail shipment car outside heavily populated areas would be accompanied by at least one escort stationed at a location on the train that would permit observation of the shipment car while in motion. Escorts would have the capability to communicate with the communications center and the cognizant local law enforcement agency.

Reporting of Safeguards Events: The Shipper of Record would notify the NRC Operations Center within one hour of discovery of credible threat (e.g., theft or significant physical damage), entry of unauthorized person, or failure/degradation in a safeguard system. Other notification time criteria would be put in place for less significant safeguards events.

H.5 Communication Considerations

In the case of extended or high visibility shipping campaigns, such as in the implementation of Management Alternative 1 or Management Alternative 3, a written public and media communications plan would be included in the overall Transportation Plan. The communication plan would generally be completed by public affairs personnel at the management sites (in cooperation with State, Tribal, and local authorities). The purpose of this plan is to ensure the exchange of accurate and timely information among the foreign research reactor operators (and their agents, if applicable), the States and Tribes, other Federal agencies, the public, and the media.

H.5.1 Public and Media Communication

The public and media communication plan for a foreign research reactor spent nuclear fuel shipment would establish public information points of contact for each agency and jurisdiction through which a shipment would pass, coordinate public education and information activities in those jurisdictions, promote communication among shipment participants by keeping State, Tribal, and local points of contact informed, and detail the procedures and sources of information in the event of an accident or other incident. To accomplish these goals, the communication plan could include the following:

Roles and Responsibilities: The roles and responsibilities of the management site, the States and Tribes, other Federal agencies, the Shipper of Record or the shipper's agent, the land carrier, the port authorities, and regional State association (if any) would be detailed.

Congressional and State Notification: DOE Headquarters Congressional and Intergovernmental Liaison Office would make appropriate Congressional notifications and respond to requests from Congressional staff and elected officials for additional information. Prior to the transport of spent nuclear fuel within or through a State, the Governor of that State would be notified in writing.

***Stakeholder, State and Tribal Notification*:** The management site, States, and Tribes generally provide for appropriate stakeholder and special interest group notification, as needed, and would respond to requests from stakeholders for additional information consistent with NRC regulations concerning safeguards.

***Media Interactions*:** Media interactions would be coordinated by the management site and would take place prior to the shipment, during the shipment, at the port of entry, after arrival at the management site, and at the completion of the shipment. These activities could range from the press briefings prior to the first shipment to a complete listing of State, Tribal, and local media print and broadcast media, to broad distribution of press releases, radio and TV news spots with accompanying factsheets, videos, and other information materials.

Public Interactions: The management site would interact with the public by answering questions, conducting briefings and meetings, and disseminating material. The States and Tribes often choose to distribute additional information packages through their emergency response organizations, civic and service organizations, local government agencies, or other special interest groups.

***Emergency Procedures*:** In the unlikely event of an accident or an incident, the management site, the Shipper of Record or its agent, and the State, Tribal, and local authorities have a coordinated approach to dealing with information about the emergency, providing timely information to the press and to the pre-established State and local contacts.

References

DOE (U.S. Department of Energy), 1995, *Program Manager's Guide to Transportation Planning*, Office of Transportation, Emergency Management, and Analytical Services of the Office of Environmental Management, Washington, DC, June.

NRC (U.S. Nuclear Regulatory Commission), 1988, *The Price-Anderson System*, NUREG BR-0079, Office of Nuclear Reactor Regulation, Washington, DC.